US-APWR DCD Revision 2
RAI Tracking Report

February 2011

Revision History

Revision	Page	Description
0	All	Original issued Including RAI responses that were submitted through October 31, 2009
1	All	Including RAI responses that were submitted through December 31, 2009
2	All	Including RAI responses that were submitted through February 28, 2010 Including editorial changes to clarify the English language and to correct typographical errors
3	All	Including RAI responses that were submitted through April 30, 2010 Including editorial changes to clarify the English language and to correct typographical errors
4	All	Including RAI responses that were submitted through August 31, 2010 Including editorial changes to clarify the English language and to correct typographical errors
5	All	Including RAI responses that were submitted through October 30, 2010 Including editorial changes to clarify the English language and to correct typographical errors
6	All	Including RAI responses that were submitted through December 31, 2010 Including editorial changes to clarify the English language and to correct typographical errors Including Conceptual Design Information as appendix.
7	All	Including RAI responses regarding seismic analysis that were submitted through December 31, 2010 Including editorial changes to clarify the English language and to correct typographical errors Including Engineering Progress to incorporate into DCD Rev. 3

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General Description

This report includes a table that identifies the impact of each response to the Request for Additional Information ("RAI") relative to the Design Control Document ("DCD") Revision 2 of US-APWR. This table shows the RAI responses which have been submitted since October 2009 and also should be incorporated into Tracking Report and DCD in future revision.

The report also includes the DCD Markups and Revision List for the RAI responses that impacted the DCD. Furthermore, the editorial changes to clarify the English language and to correct typographical errors are shown in the DCD Markups and Revision List.

Contents

For ease of using this Tracking Report, each chapter is organized in a stand alone fashion that includes a cover sheet and the following relevant information:

- DCD Revision List – a list of the revision resulting from RAI responses and others changes
- DCD Markups – a copy of the DCD pages that have changes resulting from RAI responses or others change.

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.2.1	Seismic Classification	581	03.02.01-15	2010/7/21	Y	N	N		-	DCD_03.02.01-15	4	3
		581	03.02.01-16	2010/7/21	Y	N	N		-	DCD_03.02.01-16	4	3
		581	03.02.01-17	2010/7/21	N	N	N		-	-	N/A	N/A
		581	03.02.01-18	2010/7/21	N	N	N		-	-	N/A	N/A
		684	03.02.01-19	XX/YY/2011								
		684	03.02.01-20	XX/YY/2011								
3.2.2	System Quality Group								CP RAI 67	CP_03.02.02-3	0	3
	Classification	580	03.02.02-10	2010/7/21	Y	Y	N		-	DCD_03.02.02-10	4	3
		580	03.02.02-11	2010/7/21	N	N	N		-	-	N/A	N/A
		580	03.02.02-12	2010/7/21	Y	N	N		-	DCD_03.02.02-12	4	3
		580	03.02.02-13	2010/7/21	N	N	N		-	-	N/A	N/A
		580	03.02.02-14	2010/7/21	N	N	N		-	-	N/A	N/A
		580	03.02.02-15	2010/7/21	N	N	N		-	-	N/A	N/A
		580	03.02.02-16	2010/7/21	N	N	N		-	-	N/A	N/A
		667	03.02.02-17	XX/YY/2010								
		667	03.02.02-18	XX/YY/2010								
		667	03.02.02-19	XX/YY/2010								
3.3.1	Wind Loadings											
3.3.2	Tornado Loadings											
3.4.1	Internal Flood Protection for	579	03.04.01-21	2010/5/27	Y	Y	N		-	DCD_03.04.01-21	4	3
	Onsite Equipment Failures	579	03.04.01-22	2010/5/27	Y	N	N		-	DCD_03.04.01-22	4	3
				2010/12/9	Y	N	N		-		6	3
		579	03.04.01-23	2010/5/27	Y	N	N		-	DCD_03.04.01-23	4	3
		579	03.04.01-24	2010/6/21	Y	N	N		-	DCD_03.04.01-24	4	3
		579	03.04.01-25	2010/5/27	Y	N	N		-	DCD_03.04.01-25	4	3
		579	03.04.01-26	2010/6/21	Y	N	N		-	DCD_03.04.01-26	4	3
		579	03.04.01-27	2010/6/21	Y	N	N		-	DCD_03.04.01-27	4	3
		579	03.04.01-28	2010/6/21	Y	N	N		-	DCD_03.04.01-28	4	3
3.4.2	Analysis Procedures		03.04.02-1									
			03.04.02-2									
			03.04.02-3									
			03.04.02-4									
		489	03.04.02-5	12/26/2009	N	N	N		-	-	N/A	N/A
		546	03.04.02-6	2010/4/16	N	N	N		-	-	N/A	N/A
3.5.1.1	Internally Generated Missiles											
	(Outside Containment)											
3.5.1.2	Internally-Generated Missiles											
	(Inside Containment)											
3.5.1.3	Turbine Missiles	323	03.05.01.03-3	2010/5/24	Y	N	N		-	DCD_03.05.01.03-3	4	3
3.5.1.4	Missiles Generated by											
	Tornadoes and Extreme Winds											
3.5.1.5	Site Proximity Missiles											
	(Except Aircraft)											
3.5.1.6	Aircraft Hazards											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.5.2	Components to be Protected											
	from Externally-Generated											
	Missiles											
3.5.3	Barrier Design Procedures	482	03.05.03-7	2009/12/9	N	N	N		-	-	N/A	N/A
		482	03.05.03-8	2009/12/9	Y	N	N		-	DCD_03.05.03-8	1	3
		686	03.05.03-9	XX/YY/2011								
3.6.1	Plant Design for Protection											
	Against Postulated Piping Failures											
	in Fluid Systems											
	Outside Containment											
3.6.2	Determination of Rupture Locations	459	03.06.02-20	2009/10/16	Y	N	N		-	DCD_03.06.02-20	-	2
	and Dynamic Effects Associated	459	03.06.02-21	2009/10/16	N	N	N		-	-	N\A	N\A
	with the Postulated Rupture	459	03.06.02-22	2009/10/16	Y	N	N		-	DCD_03.06.02-22	-	2
	of Piping	459	03.06.02-23	2009/10/16	Y	N	N		-	DCD_03.06.02-23	-	2
		459	03.06.02-24	2009/10/16	Y	N	N		-	DCD_03.06.02-24	-	2
		459	03.06.02-25	2009/10/16	Y	N	N		-	DCD_03.06.02-25	0	3
		459	03.06.02-26	2009/10/16	N	N	N		-	-	N\A	N\A
		459	03.06.02-27	2009/10/16	Y	N	N		-	DCD_03.06.02-27	-	2
		459	03.06.02-28	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-29	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-30	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-31	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-32	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-33	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-34	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-35	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-36	2009/10/16	N	N	N		-	-	N\A	N\A
		459	03.06.02-37	10/16/2009	Y	N	N		-	DCD_03.06.02-37	-	2
		459	03.06.02-38	10/16/2009	Y	N	N		-	DCD_03.06.02-38	-	2
		459	03.06.02-39	2009/12/1	Y	N	N		-	DCD_03.06.02-39	1	3
		636	03.06.02-40	11/24/2010	N	N	N		-	-	N\A	N\A
				12/15/2010	Y	N	N		-	DCD_03.06.02-40	7	3
		636	03.06.02-41	11/24/2010	Y	N	N		-	DCD_03.06.02-41	7	3
				12/15/2010	Y	N	N		-		7	3
		636	03.06.02-42	11/24/2010	Y	N	N		-	DCD_03.06.02-42	7	3
				12/15/2010	Y	N	N		-		7	3
		636	03.06.02-43	11/24/2010	Y	N	N		-	DCD_03.06.02-43	7	3
				12/15/2010	Y	N	N		-		7	3
		636	03.06.02-44	11/24/2010	N	N	N		-	-	N\A	N\A
				12/15/2010	Y	N	N		-	DCD_03.06.02-44	7	3
		636	03.06.02-45	11/24/2010	N	N	N		-	-	N\A	N\A
				12/15/2010	Y	N	N		-	DCD_03.06.02-44	7	3
		636	03.06.02-46	11/24/2010	N	N	N		-	-	N\A	N\A
				12/15/2010	N	N	N		-	-	N\A	N\A
		636	03.06.02-47	11/24/2010	Y	N	N		-	DCD_03.06.02-47	7	3
				12/15/2010	Y	N	N		-		7	3
		636	03.06.02-48	11/24/2010	Y	N	N		-	DCD_03.06.02-48	7	3
				12/15/2010	Y	N	N		-		7	3
3.6.3	Leak-Before-Break	485	3.6.3-18	2010/1/18	N	N	N		-	-	N/A	N/A
	Evaluation Procedures	485	3.6.3-19	2010/1/18	Y	Y	N		-	DCD_3.6.3-19	2	3
		485	3.6.3-20	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-21	2010/1/18	Y	N	N		-	DCD_3.6.3-21	2	3
		485	3.6.3-22	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-23	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-24	2010/1/18	Y	N	N		-	DCD_3.6.3-24	2	3
		485	3.6.3-25	2010/1/18	N	N	N		-	-	N/A	N/A

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.7.1	Seismic Design Parameters	494	03.07.01-2	2010/1/29	N	N	N		-	-	N/A	N/A
		494	03.07.01-3	2010/1/29	N	N	N		-	-	N/A	N/A
		494	03.07.01-4	2010/1/29	Y	Y	N		-	DCD_03.07.01-4	2	3
		602	03.07.01-5	2010/7/27	N	N	N		-	-	N/A	N/A
		643	03.07.01-6	XX/YY/2010								
		643	03.07.01-7	XX/YY/2010								
		643	03.07.01-8	XX/YY/2010								
		643	03.07.01-9	XX/YY/2010								
		643	03.07.01-10	XX/YY/2010								
		643	03.07.01-12	2010/11/11	Y	Y	N		-	DCD_03.07.01-12	6	3
		643	03.07.01-13	2010/11/11	N	N	N		-	-	N/A	N/A
		643	03.07.01-14	2010/11/11	N	N	N		-	-	N/A	N/A
		643	03.07.01-15	2010/11/11	N	N	N		-	-	N/A	N/A
		643	03.07.01-16	2010/11/11	N	N	N		-	-	N/A	N/A
		659	03.07.01-17	2010/12/28	N	N	N		-	-	N/A	N/A
		659	03.07.01-18	2010/12/28	N	N	N		-	-	N/A	N/A
3.7.2	Seismic System Analysis	495	03.07.02-2	2010/2/2	N	N	N		-	-	N/A	N/A
		212	3.7.2-3	2009/5/7	Y	N	N		-	DCD_3.7.2-3	6	3
		495	03.07.02-3A	2010/2/2	N	N	N		-	-	N/A	N/A
		495	03.07.02-4	2010/2/2	Y	N	N		-	DCD_03.07.02-4	TBD	
		495	03.07.02-5	2010/2/2	Y	N	N		-	DCD_03.07.02-5	2	3
		603	03.07.02-9	2010/7/27	N	N	N		-	-	N/A	N/A
		603	03.07.02-10	2010/8/30	N	N	N		-	-	N/A	N/A
		625	03.07.02-11/RAI 3.7.2-38	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-12/RAI 3.7.2-39	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-13/RAI 3.7.2-40	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-14/RAI 3.7.2-41	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-15/RAI 3.7.2-42	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-16/RAI 3.7.2-43	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-17/RAI 3.7.2-44	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-18/RAI 3.7.2-45	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-19/RAI 3.7.2-46	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-20/RAI 3.7.2-47	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-21/RAI 3.7.2-48	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-22/RAI 3.7.2-49	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-23/RAI 3.7.2-50	2010/11/4	N	N	N		-	-	N/A	N/A
		625	03.07.02-24/RAI 3.7.2-51	2010/11/4	N	N	N		-	-	N/A	N/A
		212	3.7.2-17	2009/5/7	N	N	N		-	-	N/A	N/A
		212	3.7.2-18	2009/5/7	N	N	N		-	-	N/A	N/A
		212	3.7.2-19	2009/5/7	N	N	N		-	-	N/A	N/A
			3.7.2-29									
			3.7.2-30									
			3.7.2-31									
			3.7.2-32									
		542	3.7.2-33	2010/3/30	N	N	N		-	-	N/A	N/A
		542	3.7.2-34	2010/3/30	N	N	N		-	-	N/A	N/A
		542	3.7.2-35	2010/3/30	N	N	N		-	-	N/A	N/A
		660	3.7.2-52	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-53	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-54	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-55	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-56	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-57	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-58	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-59	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-60	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-61	2010/12/28	N	N	N		-	-	N/A	N/A
		660	3.7.2-62	2010/12/28	N	N	N		-	-	N/A	N/A
		660	03.07.02-63	2010/12/28	Y	N	N			DCD_03.07.02-63	7	3
		660	03.07.02-64	2010/12/28	N	N	N		-		N/A	N/A
		660	03.07.02-65	2010/12/28	Y	N	N			DCD_03.07.02-65	7	3
		660	3.7.2-68	2010/12/28	N	N	N		-	-	N/A	N/A

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3.7.3	Seismic Subsystem Analysis	493	03.07.03-2	2010/1/28	Y	N	N		-	DCD_03.07.03-2	2	3
		493	03.07.03-3	2010/1/28	N	N	N		-	-	N/A	N/A
		493	03.07.03-4	2010/1/28	N	N	N		-	-	N/A	N/A
		493	03.07.03-5	2010/1/28	Y	N	N		-	DCD_03.07.03-5	2	3
		213	3.7.3-09 SUPP	2010/6/30	Y	N	N		-	DCD_3.7.3-09	4	3
3.7.4	Seismic Instrumentation											
		538	03.07.04-03	2010/3/30	Y	N	N		-	DCD_03.07.04-03	6	3
		632	03.07.04-04	2010/10/19	Y	Y	N		-	DCD_03.07.04-04	6	3
3.8.1	Concrete Containment											
		-	-	-	-	-	-	-	COL3.8(2) deleted	MAP-03-004	0	2, 3
		490	03.08.01-2	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-3	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-4	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-5	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-6	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-7	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-8	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-9	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-10	2010/2/4	N	N	N		-	-	N/A	N/A
		661	03.08.01-11	XX/YY/2010								
		661	03.08.01-12	XX/YY/2010								
		661	03.08.01-13	XX/YY/2010								
		661	03.08.01-24	2010/12/28	Y	N	N		-	DCD_03.08.01-24	7	3
		661	03.08.01-25	2010/12/28	N	N	N		-	-	N/A	N/A
		661	03.08.01-26	2010/12/28	N	N	N		-	-	N/A	N/A
3.8.3	Concrete and Steel	491	03.08.03-16	2010/3/3	N	N	N		-	-	N/A	N/A
	Internal Structures	491	03.08.03-17	2010/3/3	N	N	N		-	-	N/A	N/A
	of Steel or Concrete Containments	491	03.08.03-18	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-19	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-20	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-21	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-22	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-23	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-24	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-25	2010/3/3	N	N	N		-	-	N/A	N/A

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		662	03.08.03-26	2010/12/28	Y	N	N		-	DCD_03.08.03-26	TBD	
		662	03.08.03-27	2010/12/28	N	N	N		-	-	N/A	N/A
		662	03.08.03-28	2010/12/28	N	N	N		-	-	N/A	N/A
		662	03.08.03-29	2010/12/28	N	N	N		-	-	N/A	N/A
		662	03.08.03-30	2010/12/28	N	N	N		-	-	N/A	N/A
		662	03.08.03-31	2010/12/28	N	N	N		-	-	N/A	N/A
		662	03.08.03-32	2010/12/28	Y	N	N		-	DCD_03.08.03-32	TBD	
		676	03.08.03-33	XX/YY/2010								
		676	03.08.03-34	XX/YY/2010								
		676	03.08.03-35	XX/YY/2010								
3.8.4	Other Seismic Category I Structures	497	03.08.04-32	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-33	2010/2/19	Y	N	N		-	DCD_03.08.04-33	TBD	
		497	03.08.04-34	2010/2/19	Y	N	N		-	DCD_03.08.04-34	2	3
		497	03.08.04-35	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-36	2010/2/19	Y	N	N		-	DCD_03.08.04-36	2	3
		497	03.08.04-37	2010/2/19	N	N	N		-	-	N/A	N/A
				2011/1/27	Y	N	N		-	DCD_03.08.04-37	TBD	
		497	03.08.04-38	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-39	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-40	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-41	2010/2/19	Y	N	N		-	DCD_03.08.04-41	2	3
		497	03.08.04-42	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-43	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-44	2010/2/19	N	N	N		-	-	N/A	N/A
				2011/2/27	Y	N	N		-	DCD_03.08.04-44	TBD	
		497	03.08.04-45	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-46	2010/2/19	Y	N	N		-	DCD_03.08.04-46	2	3
		497	03.08.04-47	2010/2/19	N	N	N		-	-	N/A	N/A
		658	03.08.04-48	2010/12/28	N	N	N		-	-	N/A	N/A
		658	03.08.04-49	2010/12/28	N	N	N		-	-	N/A	N/A
3.8.5	Foundations	496	03.08.05-23	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-24	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-25	2010/2/4	Y	N	N		-	DCD_03.08.05-25	6	3
		496	03.08.05-26	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-27	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-28	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-29	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-30	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-31	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-32	2010/2/4	Y	N	N		-	DCD_03.08.05-32	2	3
		496	03.08.05-33	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-34	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-35	2010/2/4	Y	Y	N		-	DCD_03.08.05-35	6	3
		657	03.08.05-36	2010/12/28	N	N	N		-	-	N/A	N/A
		657	03.08.05-37	2010/12/28	N	N	N		-	-	N/A	N/A
		657	03.08.05-38	2010/12/28	Y	N	N		-	DCD_03.08.05-38	7	3
		657	03.08.05-39	2010/12/28	N	N	N		-	-	N/A	N/A
		657	03.08.05-40	2010/12/28	N	N	N		-	-	N/A	N/A
		657	03.08.05-41	2010/12/28	Y	Y	N		-	DCD_03.08.05-41	7	3
3.9.1	Special Topics for											
	Mechanical Components											
3.9.2	Dynamic Testing and Analysis	498	03.09.02-59	2010/1/15	N	N	N		-	-	N/A	N/A
	of Systems, Structures,	498	03.09.02-60	2010/1/15	N	N	N		-	-	N/A	N/A
	and Components	498	03.09.02-61	2010/2/3	Y	N	N		-	DCD_03.09.02-61	TBD	
		498	03.09.02-62	2010/2/3	Y	N	N		-	DCD_03.09.02-62	TBD	
		498	03.09.02-63	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-64	2010/2/3	Y	N	N		-	DCD_03.09.02-64	2	3
		498	03.09.02-65	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-66	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-67	2010/1/15	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		498	03.09.02-68	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-69	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-70	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-71	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-72	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-73	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-74	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-75	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-76	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-77	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-78	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-79	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-80	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-81	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-82	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-83	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-84	2010/2/3	N	N	N		-	-	N/A	N/A
		614	03.09.02-85	2010/9/16	N	N	N		-	-	N/A	N/A
		614	03.09.02-86	2010/9/16	N	N	N		-	-	N/A	N/A
		614	03.09.02-87	2010/9/16	N	N	N		-	-	N/A	N/A
		614	03.09.02-88	2010/9/29	N	N	N		-	-	N/A	N/A
		614	03.09.02-89	2010/9/29	Y	N	N		-	DCD_03.09.02-89	6	3
		614	03.09.02-90	2010/9/29	Y	N	N		-	DCD_03.09.02-90	6	3
		614	03.09.02-91	2010/10/28	N	N	N		-	-	N/A	N/A
		646	03.09.02-92	2010/11/11	N	N	N		-	-	N/A	N/A
				2010/12/14	N	N	N		-	-	N/A	N/A
3.9.3	ASME Code Class 1, 2, and 3											
	Components,											
	and Component Supports,											
	and Core Support Structures											
3.9.4	Control Rod Drive Systems	569	03.09.04-2	2010/5/13	Y	N	N		-	DCD_03.09.04-2	3	3
		570	03.09.04-3	2010/5/19	Y	N	N		-	DCD_03.09.04-3	4	3
		570	03.09.04-4	2010/5/19	Y	N	N		-	DCD_03.09.04-4	4	3
		570	03.09.04-5	2010/5/19	Y	N	N		-	DCD_03.09.04-5	4	3
		570	03.09.04-6	2010/5/19	N	N	N		-	-	N/A	N/A
		604	03.09.04-7	2010/7/28	Y	N	N		-	DCD_03.09.04-7	4	3
		604	03.09.04-8	2010/7/28	Y	N	N		-	DCD_03.09.04-8	4	3
		604	03.09.04-9	2010/7/28	Y	N	N		-	DCD_03.09.04-9	4	3
3.9.5	Reactor Pressure Vessel Internals	663	03.09.05-28	2011/1/21	N	N	N		-	-	N/A	N/A
		663	03.09.05-29	2011/1/21	N	N	N		-	-	N/A	N/A
		663	03.09.05-30	2011/1/21	Y	N	N		-	DCD_03.09.05-30	TBD	
		663	03.09.05-31	2011/1/21	N	N	N		-	-	N/A	N/A
		663	03.09.05-32	2011/1/21	Y	N	N		-	DCD_03.09.05-32	TBD	
		663	03.09.05-33	2011/1/21	Y	N	N		-	DCD_03.09.05-33	TBD	
		663	03.09.05-34	2011/1/21	N	N	N		-	-	N/A	N/A
3.9.6	Functional Design, Qualification,											
	and Inservice Testing Programs											
	for Pumps, Valves,											
	and Dynamic Restraints											
3.10	Seismic/Dynamic Qual	486	03.10-10	2009/12/9	N	N	N		-	-	N/A	N/A
	of Mech/Elec Eqmt	486	03.10-11	2009/12/9	Y	N	N		-	DCD_03.10-11	1	3
		486	03.10-12	2009/12/9	Y	N	N		-	DCD_03.10-12	1	3
		486	03.10-13	2009/12/25	Y	N	N		-	DCD_03.10-13	1	3
		486	03.10-14	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-15	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-16	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-17	2009/12/25	N	N	N		-	-	N/A	N/A
3.11	Environmental Qual	445	03.11-16	2009/9/29	Y	Y 1.8	N		-	DCD_03.11-16	0	3

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
	of Mech/Elec Eqmt	511	03.11-17	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-17 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-18	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-18 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-19	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-19 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-20	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-20 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-21	2010/2/2	Y	N	N		-	DCD_03.11-21	2	3
		511	03.11-21 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-22	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-22 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-23	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-23 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-24	2010/2/2	Y	N	N		-	DCD_03.11-24	2	3
		511	03.11-25	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-25 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-26	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-26 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		511	03.11-27	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-28	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-28 Supp	2010/6/28	N	N	N		-	-	N/A	N/A
		512	03.11-29	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-30	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-31	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-32	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-33	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-34	2010/1/28	Y	N	N		-	DCD_03.11-34	4	3
		512	03.11-35	2010/1/28	N	N	N		-	-	N/A	N/A
		589	03.11-36	2010/7/8	N	N	N		-	-	N/A	N/A
		589	03.11-37	2010/7/8	N	N	N		-	-	N/A	N/A
		589	03.11-38	2010/7/8	Y	N	N		-	DCD_03.11-38	4	3
		650	03.11-39	XX/YY/2010								
		650	03.11-40	XX/YY/2010								
3.12	ASME Code Class 1, 2, and 3	465	03.12-17	2009/12/2	Y	N	N		-	DCD_O3.12-17	1	3
	Piping Systems,	465	03.12-18	2009/11/18	N	N	N		-	-	N/A	N/A
	Piping Components	465	03.12-19	2009/11/18	Y	N	N		-	DCD_03.12-19	0	3
	and their Associated Supports	465	03.12-20	2009/11/18	Y	N	N		-	DCD_03.12-20	0	3
		465	03.12-21	2009/11/18	N	N	N		-	-	N/A	N/A
		465	03.12-22	2009/11/18	N	N	N		-	-	N/A	N/A
		465	03.12-23	2009/12/2	Y	N	N		-	DCD_03.12-23	1	3
		465	03.12-24	2009/11/18	Y	N	N		-	DCD_03.12-24	0	3
3.13	Threaded Fasteners -	273	3.13-1	2009/4/9	Y	N	N		-	DCD_3.13-1	3	2
	ASME Code Class 1, 2, and 3	273	3.13-2	2009/4/9	Y	N	N		-	DCD_3.13-2	3	2
		273	3.13-3	2009/4/9	Y	N	N		-	DCD_3.13-3	3	2
		273	3.13-4	2009/4/9	Y	N	N		-	DCD_3.13-4	3	2
		273	3.13-5	2009/4/9	Y	N	N		-	DCD_3.13-5	3	2
		-	-	-	-	-	-	-	COL3.13(1) deleted	MAP-03-015	-	2
		-	-	-	-	-	-	-	COL3.13(2) deleted	MAP-03-016	-	2
3.13	Threaded Fasteners -											
	ASME Code Class 1, 2, and 3											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	264	05.02.01.01-1	2009/10/2	Y	Y	N		-	DCD_05.02.01.01-1	-	2
				2009/12/15	Y	Y	N		-	DCD_05.02.01.01-1	7	3
5.2.1.2	Applicable Code Cases	575	05.02.01.02-7	2010/5/7	Y	N	N		-	DCD_05.02.01.02-7	7	3
5.2.2	Overpressure Protection											
5.2.3	Reactor Coolant Pressure Boundary Materials	289	05.02.03-12	2010/3/1	Y	N	N		-	DCD_O5.02.03-12	7	3
		509	05.02.03-18	2010/1/29	N	N	N		-	-	N/A	N/A
		540	05.02.03-19	2010/6/4	Y	N	N		-	DCD_O5.02.03-19	7	3
		540	05.02.03-20	2010/6/4	N	N	N		-	-	N/A	N/A
		540	05.02.03-21	2010/6/4	Y	N	N		-	DCD_O5.02.03-21	7	3
		540	05.02.03-22	2010/6/4	Y	N	N		-	DCD_O5.02.03-22	7	3
		540	05.02.03-23	2010/6/4	Y	N	N		-	DCD_O5.02.03-23	7	3
		540	05.02.03-24	2010/6/4	N	N	N		-	-	N/A	N/A
		540	05.02.03-25	2010/6/4	N	N	N		-	-	N/A	N/A
		-	-	-	-	-	-	-	COL 5.2(4) revised	MAP-05-001	TBD	
		-	-	-	-	-	-	-	COL 5.2(5) revised	MAP-05-002	TBD	
		644	05.02.03-26	2010/11/8	Y	N	N		-	DCD_05.02.03-26	7	3
		644	05.02.03-27	2010/11/8	Y	N	N		-	DCD_05.02.03-27	7	3
				2011/2/20	Y	N	N		-	DCD_05.02.03-27	TBD	
		644	05.02.03-28	2010/11/8	Y	N	N		-	DCD_05.02.03-28	7	3
		644	05.02.03-29	2010/11/8	Y	N	N		-	DCD_05.02.03-29	7	3
		644	05.02.03-30	2010/11/8	N	N	N		-	-	N/A	N/A
				2011/2/20	N	N	N		-	-	N/A	N/A
		644	05.02.03-31	2010/11/8	N	N	N		-	-	N/A	N/A
				2011/2/20	N	N	N		-	-	N/A	N/A
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	254	05.02.04-8	2009/4/17	Y	N	N		-	DCD_05.02.04.-8	3	2
				2009/10/2	Y	Y	N		-	DCD_05.02.04-8	-	2
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	478	05.02.05-11	2009/12/2	Y	N	N		-	DCD_05.02.05-11	1	3
		549	05.02.05-12	2010/4/9	Y	N	N		-	DCD_05.02.05-12	3	3
5.3.1	Reactor Vessel Materials											
5.3.2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	588	05.03.02-2	2010/6/14	N	N	N		-	-	N/A	N/A
		588	05.03.02-3	2010/6/14	N	N	N		-	-	N/A	N/A
		588	05.03.02-4	2010/6/14	N	N	N		-	-	N/A	N/A
		588	05.03.02-5	2010/6/14	Y	N	N		-	DCD_05.03.02-5	4	3
		588	05.03.02-6	2010/6/14	N	N	N		-	-	N/A	N/A
		588	05.03.02-7	2010/6/14	N	N	N		-	-	N/A	N/A
		588	05.03.02-8	2010/6/14	N	N	N		-	-	N/A	N/A
5.3.3	Reactor Vessel Integrity											
5.4	Reactor Coolant System Component and Subsystem Design											
5.4.1.1	Pump Flywheel Integrity (PWR)											
5.4.2.1	Steam Generator Materials											
5.4.2.2	Steam Generator Program											

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5.4.7	Residual Heat Removal (RHR)	548	05.04.07-12	2010/4/6	Y	N	N		-	DCD_05.04.07-12	3	3
	System	617	05.04.07-13	2010/9/14	N	N	N		-	-	N/A	N/A
5.4.11	Pressurizer Relief Tank											
5.4.12	Reactor Coolant System	OI	05.04.12-1	2009/10/2	N	N	N		-	-	N/A	N/A
	High Point Vents											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.1.1	Criticality Safety of Fresh and	155	09.01.01.-1	2009/2/10	N	N	N		-	-	N/A	N/A
	Spent Fuel Storage and Handling	155	09.01.01-2	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-3	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-4	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-5	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-6	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-7	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-8	2009/2/10	N	N	N		-	-	N/A	N/A
		247	09.01.01-9	2009/3/30	Y	N	N		-	DCD_09.01.01-9	-	2
		247	09.01.01-10	2009/3/30	Y	Y	N		-	DCD_09.01.01-10	-	2
		382	09.01.01-11	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-12	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-13	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-14	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-15	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-16	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-17	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-18	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-19	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-20	2009/7/7	N	N	N		-	-	N/A	N/A
		382	09.01.01-21	2009/7/7	N	N	N		-	-	N/A	N/A
		647	09.01.01-22	2010/11/11	N	N	N		-	-	N/A	N/A
		647	09.01.01-23	2010/11/11	N	N	N		-	-	N/A	N/A
9.1.1	Criticality Safety of Fresh and											
	Spent Fuel Storage and Handling											
9.1.2	New and Spent Fuel Storage											
9.1.3	Spent Fuel Pool Cooling and											
	Cleanup System											
9.1.4	Light Load Handling System	507	09.01.04-16	2010/2/15	Y	N	N		-	DCD_09.01.04-16	2	3
	(Related to Refueling)	555	09.01.04-17	2010/6/4	Y	N	N		-			
				2010/6/16	Y	N	N		-	DCD_09.01.04-17	4	3
		555	09.01.04-18	2010/6/4	Y	N	N		-			
				2010/6/16	Y	N	N		-	DCD_09.01.04-18	4	3
		555	09.01.04-19	2010/6/4	Y	N	N		-			
				2010/6/16	Y	N	N		-	DCD_09.01.04-19	4	3
		555	09.01.04-20	2010/6/4	Y	N	N		-			
				2010/6/16	Y	N	N		-	DCD_09.01.04-20	4	3
		633	09.01.04-21	2010/10/21	Y	N	N		-	DCD_09.01.04-21	5	3
9.1.5	Overhead Heavy Load	563	09.01.05-14	2010/6/15	Y	N	N		-	DCD_09.01.05-14	4	3
	Handling Systems	563	09.01.05-15	2010/6/15	Y	N	N		-	DCD_09.01.05-15	4	3
		563	09.01.05-16	2010/6/15	Y	N	N		-	DCD_09.01.05-16	4	3
		563	09.01.05-17	2010/6/15	Y	N	N		-	DCD_09.01.05-17	4	3
		616	09.01.05-18	2010/9/22	Y	N	N		-	DCD_09.01.05-18	5	3
		616	09.01.05-19	2010/9/22	N	N	N		-	-	N/A	N/A
9.2.1	Station Service Water System	585	09.02.01-32	2010/9/24	Y	Y	N		-	DCD_09.02.01-32	6	3
		585	09.02.01-33	2010/9/24	Y	Y	N		-	DCD_09.02.01-33	6	3
		585	09.02.01-34	2010/9/24	Y	Y	N		-	DCD_09.02.01-34	6	3
		585	09.02.01-35	2010/9/24	Y	Y	N		-	DCD_09.02.01-35	6	3
		585	09.02.01-36	2010/9/24	Y	Y	N		-	DCD_09.02.01-36	6	3
		585	09.02.01-37	2010/9/24	Y	Y	N		-	DCD_09.02.01-37	6	3
		585	09.02.01-38	2010/9/24	Y	Y	N		-	DCD_09.02.01-38	6	3
		585	09.02.01-39	2010/9/24	Y	Y	N		-	DCD_09.02.01-39	6	3
		585	09.02.01-40	2010/9/24	Y	N	N		-	DCD_09.02.01-40	6	3
		585	09.02.01-41	2010/9/24	Y	Y	N		-	DCD_09.02.01-41	6	3
		585	09.02.01-42	2010/9/24	Y	N	N		-	DCD_09.02.01-42	6	3
		585	09.02.01-43	2010/9/24	Y	Y	N		-	DCD_09.02.01-43	6	3
		585	09.02.01-44	2010/9/24	Y	N	N		-	DCD_09.02.01-44	6	3
		585	09.02.01-45	2010/9/24	Y	Y	N		-	DCD_09.02.01-45	6	3
		585	09.02.01-46	2010/9/24	Y	Y	N		-	DCD_09.02.01-46	6	3

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		585	09.02.01-47	2010/9/24	N	N	N		-	-	N/A	N/A
		585	09.02.01-48	2010/9/24	N	N	N		-	-	N/A	N/A
		585	09.02.01-49	2010/9/24	Y	Y	N		-	DCD_09.02.01-49	6	3
		585	09.02.01-50	2010/9/24	N	N	N		-	-	N/A	
		585	09.02.01-51	2010/9/24	Y	Y	N		-	DCD_09.02.01-51	6	3
		585	09.02.01-52	2010/9/24	Y	Y	N		-	DCD_09.02.01-52	6	3
		585	09.02.01-53	2010/9/24	N	N	N		-	-	N/A	N/A
		585	09.02.01-54	2010/9/24	N	N	N		-	-	N/A	N/A
		585	09.02.01-55	2010/9/24	Y	Y	N		-	DCD_09.02.01-55	6	3
		585	09.02.01-56	2010/9/24	N	N	N		-	-	N/A	N/A
		585	09.02.01-57	2010/9/24	Y	Y	N		-	DCD_09.02.01-57	6	3
		585	09.02.01-58	2010/9/24	Y	Y	N		-	DCD_09.02.01-58	6	3
		585	09.02.01-59	2010/9/24	Y	Y	N		-	DCD_09.02.01-59	6	3
9.2.2	Reactor Auxiliary	567	09.02.02-46	2010/5/7	Y	N	N		-	DCD_09.02.02-46	3	3
	Cooling Water Systems	571	09.02.02-47	2010/6/8	N	N	N		-	-	N/A	N/A
		571	09.02.02-48	2010/6/8	Y	N	N		-	DCD_09.02.02-48	TBD	
		571	09.02.02-49	2010/6/8	N	N	N		-	-	N/A	N/A
		571	09.02.02-50	2010/6/8	Y	N	N		-	DCD_09.02.02-50	TBD	
		571	09.02.02-51	2010/6/8	Y	N	N		-	DCD_09.02.02-51	TBD	
		571	09.02.02-52	2010/6/8	Y	N	N		-	DCD_09.02.02-52	TBD	
		571	09.02.02-53	2010/6/8	Y	N	N		-	DCD_09.02.02-53	TBD	
		571	09.02.02-54	2010/6/8	N	N	N		-	-	N/A	N/A
		571	09.02.02-55	2010/6/8	Y	N	N		-	DCD_09.02.02-55	TBD	
		571	09.02.02-56	2010/6/8	Y	N	N		-	DCD_09.02.02-56	TBD	
		571	09.02.02-57	2010/6/8	Y	N	N		-	DCD_09.02.02-57	TBD	
		571	09.02.02-58	2010/6/8	Y	N	N		-	DCD_09.02.02-58	TBD	
		571	09.02.02-59	2010/6/8	Y	N	N		-	DCD_09.02.02-59	TBD	
		571	09.02.02-60	2010/6/8	N	N	N		-	-	N/A	N/A
		571	09.02.02-61	2010/6/8	N	N	N		-	-	N/A	N/A
		571	09.02.02-62	2010/6/8	Y	N	N		-	DCD_09.02.02-62	TBD	
		571	09.02.02-63	2010/6/8	Y	N	N		-	DCD_09.02.02-63	TBD	
		571	09.02.02-64	2010/6/8	Y	N	N		-	DCD_09.02.02-64	TBD	
		571	09.02.02-65	2010/6/8	Y	N	N		-	DCD_09.02.02-65	TBD	
		571	09.02.02-66	2010/6/8	Y	N	N		-	DCD_09.02.02-66	TBD	
		571	09.02.02-67	2010/6/8	Y	N	N		-	DCD_09.02.02-67	TBD	
		571	09.02.02-68	2010/6/8	Y	N	N		-	DCD_09.02.02-68	TBD	
		576	09.02.02-69	2010/6/8	N	N	N		-	-	N/A	N/A
		584	09.02.02-70	2010/6/10	Y	N	N		-	DCD_09.02.02-70	TBD	
		584	09.02.02-71	2010/6/10	Y	N	N		-	DCD_09.02.02-71	TBD	
		584	09.02.02-72	2010/6/10	Y	N	N		-	DCD_09.02.02-72	TBD	
		584	09.02.02-73	2010/6/10	N	N	N		-	-	N/A	N/A
		584	09.02.02-74	2010/6/10	N	N	N		-	-	N/A	N/A
		584	09.02.02-75	2010/6/10	N	N	N		-	-	N/A	N/A
		584	09.02.02-76	2010/6/10	Y	N	N		-	DCD_09.02.02-76	TBD	
		584	09.02.02-77	2010/6/10	Y	N	N		-	DCD_09.02.02-77	TBD	
		584	09.02.02-78	2010/6/10	Y	N	N		-	DCD_09.02.02-78	TBD	
		584	09.02.02-79	2010/6/10	N	N	N		-	-	N/A	N/A
9.2.4	Potable and Sanitary Water Systems											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.2.5	Ultimate Heat Sink	286	09.02.05-1	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-1	4	3
		286	09.02.05-2	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-2	4	3
		286	09.02.05-3	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-3	4	3
		286	09.02.05-4	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-4	4	3
		286	09.02.05-5	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	N	N	N		-	-	N/A	N/A
		286	09.02.05-6	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-6	4	3
		286	09.02.05-7	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-7	4	3
		286	09.02.05-8	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-8	4	3
		286	09.02.05-9	2009/5/12	N	N	N		-	-	N/A	N/A
				2010/7/7	Y	N	N		-	DCD_09.02.05-9	4	3
9.2.6	Condensate Storage Facilities											
9.3.1	Compressed Air System											
9.3.2	Process and Post-accident	461	09.03.02-12	2009/11/17	Y	N	N		-	DCD_09.03.02-12	1	3
	Sampling Systems	526	09.03.02-13	2010/4/7	Y	Y	N		-	DCD_09.03.02-13	3	3
		526	09.03.02-14	2010/4/7	N	N	N		-	-	N/A	N/A
		526	09.03.02-15	2010/4/7	Y	N	N		-	DCD_09.03.02-15	3	3
		526	09.03.02-16	2010/4/7	N	N	N		-	-	N/A	N/A
9.3.3	Equipment and Floor	426	09.03.03-15	2009/9/14	Y	N	N		-	DCD_09.03.03-15	-	2
	Drainage System	426	09.03.03-16	2009/9/14	Y	N	N		-	DCD_09.03.03-16	0	3
		426	09.03.03-17	2009/9/14	Y	N	N		-	DCD_09.03.03-17	-	2
		591	09.03.03-18	2010/7/7	N	N	N		-	-	N/A	N/A
		591	09.03.03-19	2010/7/7	Y	N	N		-	DCD_09.03.03-19	4	3
9.3.4	Chemical and Volume Control Syst	384	09.03.04-10	2009/7/17	N	N	N		-	-	N/A	N/A
	(PWR)			2010/4/7	Y	Y	N		-	DCD_09.03.04-10	3	3
	(Including Boron Recovery System)											
9.4.1	Control Room Area	63	09.04.01-14	2008/10/3	N	N	N	fin.	-			
	Ventilation System			2010/6/29	N	N	N		-	-	N/A	N/A
		327	09.04.01-5	2009/6/19	N	N	N		-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		327	09.04.01-9	2010/1/29	Y	N	N		-	DCD_09.04.01-9	2	3
		475	09.04.01-12A	2009/11/20	Y	Y	N		-	DCD_09.04.01-12A	1	3
		475	09.04.01-13A	2009/11/20	Y	N	N		-	DCD_09.04.01-13A	1	3
		475	09.04.01-14A	2009/11/20	N	N	N		-	-	N/A	N/A
		484	09.04.01-15A	2009/12/9	N	N	N		-	-	N/A	N/A
		582	09.04.01-16	2010/7/16	Y	N	N		-	DCD_09.04.01-16	4	3
		582	09.04.01-17	2010/7/16	Y	N	N		-	DCD_09.04.01-17	4	3
		582	09.04.01-18	2010/7/16	N	N	N		-	-	N/A	N/A
		582	09.04.01-19	2010/7/16	N	N	N		-	-	N/A	N/A
		582	09.04.01-20	2010/7/16	Y	N	N		-	DCD_09.04.01-20	4	3
		582	09.04.01-21	2010/7/16	Y	N	N		-	DCD_09.04.01-21	4	3
		582	09.04.01-22	2010/7/16	Y	N	N		-	DCD_09.04.01-22	4	3
		582	09.04.01-23	2010/7/16	N	N	N		-	-	N/A	N/A
		642	09.04.01-24	2010/11/5	Y	N	N		-	DCD_09.04.01-24	5	3
9.4.2	Spent Fuel Pool Area	539	09.04.02-4	2010/4/1	Y	N	N		-	DCD_09.04.02-4	3	3
	Ventilation System	539	09.04.02-5	2010/4/1	Y	N	N		-	DCD_09.04.02-5	3	3

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		592	09.04.02-6	2010/7/7	Y	N	N		-	DCD_09.04.02-6	4	3
9.4.3	Auxiliary and Radwaste Area	483	09.04.03-08	2010/2/5	Y	N	N		-	DCD_09.04.03-08	2	3
	Ventilation System	483	09.04.03-09	2010/2/5	Y	N	N		-	DCD_09.04.03-09	2	3
		483	09.04.03-10	2010/2/5	N	N	N		-	-	N/A	N/A
		634	09.04.03-11	2010/10/15	Y	N	N		-	DCD_09.04.03-11	5	3
		634	09.04.03-12	2010/10/15	Y	N	N		-	DCD_09.04.03-12	5	3
		634	09.04.03-13	2010/10/15	Y	N	N		-	DCD_09.04.03-13	5	3
9.4.4	Turbine Area Ventilation System	541	09.04.03-4	2010/3/30	N	N	N		-	-	N/A	N/A
		541	09.04.04-5	2010/3/30	Y	N	N		-	DCD_09.04.04-5	3	3
		586	09.04.04-6	2010/6/10	N	N	N		-	-	N/A	N/A
9.4.5	Engineered Safety Feature	64	09.04.05-1/9.4.5-3	2008/10/6	N	N	N	fin.	-			
	Ventilation System			2010/6/29	N	N	N		-	-	N/A	N/A
		64	09.04.05-1/9.4.5-4	2008/10/6	N	N	N	fin.	-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		64	09.04.05-1/9.4.5-22	2008/10/6	N	N	N	fin.	-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		356	09.04.05-3	2009/7/17	N	N	N		-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		356	09.04.05-4	2009/7/17	N	N	N		-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		356	09.04.05-9	2009/7/17	N	N	N		-			
				2010/6/29	N	N	N		-	-	N/A	N/A
		474	09.04.05-10	11/13/2009	Y	N	N		-	DCD_09.04.05-10	0	3
		583	09.04.05-11	2010/6/22	Y	N	N		-	DCD_09.04.05-11	4	3
		583	09.04.05-12	2010/6/22	Y	N	N		-	DCD_09.04.05-12	4	3
		666	09.04.05-13	2010/12/20	N	N	N		-	-	N/A	N/A
		670	09.04.05-14	2010/12/28	Y	N	N		-	DCD_09.04.05-14	7	3
		670	09.04.05-15	2010/12/28	N	N	N		-	-	N/A	N/A
		670	09.04.05-16	2010/12/28	Y	N	N		-	DCD_09.04.05-16	7	3
		670	09.04.05-17	2010/12/28	Y	N	N		-	DCD_09.04.05-17	7	3
		670	09.04.05-18	2010/12/28	N	N	N		-	-	N/A	N/A
9.5.1	Fire Protection Program	537	09.05.01-18	04/13/2010	Y	N	N		-	DCD_09.05.01-18	3	3
		537	09.05.01-19	04/13/2010	Y	N	N		-	DCD_09.05.01-19	3	3
9.5.2	Communications Systems											
9.5.3	Lighting Systems											
9.5.4	Emergency Diesel Engine Fuel	467	09.05.04-43	11/10/2009	Y	Y	N		-	DCD_09.05.04-43	1	3
	Oil Storage and Transfer System	468	09.05.04-44	2009/12/10	Y	Y	N		-	DCD_09.05.04-44	1	3
		468	09.05.04-45	2009/12/10	Y	N	N		-	DCD_09.05.04-45	1	3
		468	09.05.04-46	2009/12/10	Y	N	N		-	DCD_09.05.04-46	1	3
		468	09.05.04-47	2009/12/10	Y	N	N		-	DCD_09.05.04-47	1	3
		468	09.05.04-48	2009/12/10	Y	N	N		-	DCD_09.05.04-48	1	3
		468	09.05.04-49	2009/12/10	N	N	N		-	-	N/A	N/A
		565	09.05.04-50	2010/6/15	Y	N	N		-	DCD_09.05.04-50	4	3
		565	09.05.04-51	2010/6/15	Y	N	N		-	DCD_09.05.04-51	4	3
9.5.5	Emergency Diesel Engine											
	Cooling Water System											
9.5.6	Emergency Diesel Engine											
	Starting System	504	09.05.06-24	12/23/09	Y	N	N		-	DCD_09.05.06-24	1	3
		504	09.05.06-25	12/23/09	Y	N	N		-	DCD_09.05.06-25	1	3

Chapter:9

SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.5.7	Emergency Diesel Engine	469	09.05.07-18	11/6/2009	N	N	N		-	-	N/A	N/A
	Lubrication System	469	09.05.07-19	11/6/2009	N	N	N		-	-	N/A	N/A
		506	09.05.07-20	2010/1/29	Y	N	N		-	DCD_09.05.07-20	2	3
		506	09.05.07-21	2010/1/29	N	N	N		-	-	N/A	N/A
		506	09.05.07-22	2010/1/29	Y	N	N		-	DCD_09.05.07-22	2	3
		506	09.05.07-23	2010/1/29	Y	N	N		-	DCD_09.05.07-23	2	3
		556	09.05.07-24	2010/4/27	Y	N	N		-	DCD_09.05.07-24	3	3
9.5.8	Emergency Diesel Engine	470	09.05.08-18	2009/12/2	Y	N	N		-	DCD_09.05.08-18	1	3
	Combustion Air Intake and	470	09.05.08-19	2009/12/2	N	N	N		-	-	N/A	N/A
	Exhaust System	470	09.05.08-20	2009/12/2	Y	N	N		-	DCD_09.05.08-20	1	3
		470	09.05.08-21	2009/12/2	Y	N	N		-	DCD_09.05.08-21	1	3
		470	09.05.08-22	2009/12/2	Y	N	N		-	DCD_09.05.08-22	1	3
		505	09.05.08-23	2010/2/1	N	N	N		-	-	N/A	N/A
		505	09.05.08-24	2010/2/1	N	N	N		-	-	N/A	N/A
		505	09.05.08-25	2010/2/1	Y	N	N		-	DCD_09.05.08-25	2	3
		557	09.05.08-26	2010/6/14	Y	N	N		-	DCD_09.05.08-26	5	3
		618	09.05.08-27	2010/11/4	Y	N	N		-	DCD_09.05.08-27	5	3

Chapter 1

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
1-xiv	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “CTS.”
1-xvi	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “ECS.”
1-xvii	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “FDS.”
1-xviii	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “FSS” and “FTS.”
1-xxi	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “LMS.”
1-xxvi	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “PWS.”
1-xxxi	Acronyms	Acronyms were reviewed due to the adjustment with System code. Delete “VDS.”
1.2-8	Section 1.2.1.2.6	Use the acronym “RPS.”
1.2-29	Section 1.2.1.5.4.1 4th line from top	Change “basks” to “baskets.”
1.2-43	Section 1.2.1.5.5 5th line from top	Use the acronym “RPS.”

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	**Description of Change**
1.8-14	Table 1.8-2 Sheet 9 COL 3.7(25)	Change: "*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure utilizing the program ACS-SASSI SSI Version 2.2 (Reference 3.7-17) which contains time history input incoherence function capability.*" to "*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS SASSI (Reference 3.7-17) which contains time history input incoherence function capability.*" Reason: Fulfills commitment to provide site-specific SSI analysis for PS/B. [RAI 495, Question 3.7.2-4]
1.8-17	Table 1.8-2 Sheet 8 COL 3.8(30)	Add new COL item: "3.8(30) *When a coefficient of friction of 0.7 is used in calculating sliding resistance F_s, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required."* Reason: If a coefficient of friction of less than 0.7 is justified by the COL Applicant, "roughening" of the concrete is not required. [RAI 657-5135, Question 03.08.05-41]
1.9-3	Table 1.9.1-1 Sheet 1 of 15 RG 1.7	Correct the title of RG 1.7.
1.9-4	Table 1.9.1-1 Sheet 2 of 15 RG 1.16	Reflect to be withdrawn according to the latest status of regulation.
1.9-6	Table 1.9.1-1 Sheet 4 of 15 RG 1.50	Change the status "no exceptions" to "exceptions." Reason: The post-welded baking may be used in lieu of maintaining preheat until post weld heat treatment is performed. [RAI 644-5077, Question 05.02.03-27]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
1.9-8	Table 1.9.1-1 Sheet 6 of 15 RG 1.83	Reflect to be withdrawn according to the latest status of regulation.
1.9-11	Table 1.9.1-1 Sheet 9 of 15 RG 1.131	Reflect to be withdrawn according to the latest status of regulation.
1.9-12	Table 1.9.1-1 Sheet 10 of 15 RG 1.135, 1.139, 1.148	Reflect to be withdrawn according to the latest status of regulation.
1.9-16	Table 1.9.1-1 Sheet 14 of 15 RG 1.193	Editorial correction: Change the status “Not applicable” to “Conformance with no exceptions identified.”
1.9-18	Table 1.9.1-2 Sheet 1 of 2 RG 4.5, 4.6, 4.8	Reflect to be withdrawn according to the latest status of regulation.
1.9-26	Table 1.9.1-4 Sheet 1 of 4 RG 8.6	Reflect to be withdrawn according to the latest status of regulation.
1.9-128	Table 1.9.2-6 Sheet 25 of 30 SRP 6.6	Change the status: “Conformance with exceptions. Criteria 8,9,10 and 11: There are discussed in inservice inspection program prepared for COL application.” To “Conformance with exceptions. Criteria 11: This is a COL application responsibility.” Change the “Appears in DCD Chapter/Section” 6.5.2 to 6.6. Reason: Criteria 8, 9 and 10 have been addressed in Revision 2 of DCD, Section 6.6.1, last paragraph.
1.9-148	Table 1.9.2-7 Sheet 15 of 19 BTP 7-14	Editorial correction: Add 7.9.2.2 and 7.9.2.6 in “Appears in DCD Chapter/Section.”

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
1.9-177	Table 1.9.2-9 Sheet 10 of 30 SRP 9.2.6	Editorial correction: Change the status: “US-APWR design does not have a condensate storage system.” to “Condensate Storage Facilities has no safety related functions. US-APWR is not multiple unit.” Change the Appears in DCD Chapter/Section “N/A” to “9.2.6.”

ACRONYMS AND ABBREVIATIONS (Continued)

CI	containment isolation
CIS	containment internal structure
CIV	containment isolation valve
CMT	chemical mixing tank
CMTR	Certified Material test Report
COC	certificate of compliance
COL	Combined License
COLA	Combined License Application
COLR	core operating limits report
COT	channel operational test
CPET	containment phenomenological event tree
CPG	containment performance goal
CPS	condensate polishing system
CPU	central processing unit
Cr	chromium
CRDM	control rod drive mechanism
CRDMCS	control rod drive mechanism control system
CRDS	control rod drive system
CRE	control room envelope
CRHS	control room habitability system
CRMP	configuration risk management program
CS	containment spray
CS/RHR	containment spray/residual heat removal
CS/RHRS	containment spray/residual heat removal system
CSA	channel statistical accuracy
CSDRS	certified seismic design response spectra
CSET	containment system event tree
CSNI	Committee on the Safety of Nuclear Installations
CSS	containment spray system
CSTF	condensate storage and transfer facilities
CT	compact tension
~~CTS~~	~~condenser tube cleaning equipment~~
CTW	cooling tower
CV	control valve
CVCS	chemical and volume control system
CVDP	C/V reactor coolant drain pump
CVDT	containment vessel reactor coolant drain tank

ACRONYMS AND ABBREVIATIONS (Continued)

DSS	digital safety system
DTM	design team manager
DV	depressurization valve
DVI	direct vessel injection
DWS	demineralized water system
DWTSS	demineralized water transfer and storage system
E/O	electrical to optical (or optical to electrical)
EAB	exclusion area boundary
EAC/PSS	emergency ac power supply system
EARWS	evacuation alarm and remote warning system
ECC	emergency core cooling
ECCS	emergency core cooling system
ECOM	error of commission
ECP	electrical corrosion potential
~~ECS~~	~~emergency communications system~~
ECWS	essential chilled water system
EDE	effective dose equivalent
EDS	equipment drain system
EF	error factor
EFPD	effective full power days
EFW	emergency feedwater
EFWPAVS	emergency feedwater pump area HVAC system
EFWS	emergency feedwater system
EH/C	electric heating coil
EHGS	turbine electro-hydraulic governor control system
EIA	Energy Information Administration
EIF	electrical interface system
~~ELS~~	~~emergency letdown system~~
EMI	electromagnetic interference
EOC	end-of-cycle
EOF	emergency operations facility
EOL	end-of-life
EOM	error of omission
EOP	emergency operating procedure
EOST	electrical overspeed trip device
EPA	containment electric penetration assembly
EPG	emergency procedure guideline

ACRONYMS AND ABBREVIATIONS (Continued)

EPRI	Electric Power Research Institute
EPS	emergency power source
EQ	environmental qualification
EQDP	equipment qualification data package
EQSDS	equipment qualification summary data sheet
ERAC	electrical rigid aluminum conduit
ERDA	Energy Research and Development Administration (now U.S. DOE)
ERDS	emergency response data system
ERSC	electrical rigid steel conduit
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESFVS	engineered safety features ventilation system
ESLS	electrical system logic system
ESQDSR	Equipment Qualification Data Summary Report
ESQR	Equipment Seismic Qualification Report
ESW	essential service water
ESWP	essential service water pump
ESWPT	essential service water pipe tunnel
ESWS	essential service water system
ESX	ex-vessel steam explosion
ET	event tree
ETSB	effluent treatment system branch
EV	elevartor
EZB	exclusion zone boundary
FA	function allocation
FAB	feed and bleed
FAC	flow-accelerated corrosion
FATT	fracture appearance transit temperature
FCC	Federal Communications Commission
FCV	feedwater control valve
~~FDS~~	~~fire detection systems~~
FE	finite element
Fe	iron
FEM	finite element method
FHA	fire hazard analysis
FHS	fuel handling system
FIRS	foundation input response spectra

ACRONYMS AND ABBREVIATIONS (Continued)

FLB	feedwater line break
FLML	failure to maintain water level
FMEA	failure modes and effects analysis
FOS	fuel oil storage and transfer system
FP	fission product
FPP	fire protection program
FPS	fire protection system
FRA	functional requirements analysis
FS	fuel system
FSAR	Final Safety Analysis Report
FSHS	fuel storage and handling system
~~FSS~~	~~fire protection water supply system~~
FT	fault tree
~~FTS~~	~~fuel transfer system~~
FV	Fussell Vesely
FVW	Fussell Vesely worth
FWLB	feedwater line break
FWS	feedwater system
g	gravity
GA	general arrangement
Gd2O3	gadolinia
GDC	General Design Criteria
GFO	governor free operation
GLBS	generator load break switch
GMAW	gas metal arc welding
GMRS	ground motion response spectra
GOMS	goals, operators, methods, and selection
GSS	gland seal system
GT/B	gas turbine building
GT/GS	gas turbine generator system
GTAW	gas tungsten arc welding
GTG	gas turbine generator
GTPS	generator transformer protection system
GWMS	gaseous waste management system
HA	human action
HAZ	heat-affected zone
HCI	hydrochloric acid

ACRONYMS AND ABBREVIATIONS (Continued)

LB	lower bound
LBB	leak before break
LBLOCA	large break loss of coolant accident
LCO	limiting condition for operation
LCS	local control station
LD	low dependence
LDP	large display panel
LER	licensee event report
LERF	large early release frequency
LHR	linear heat rate
LHSI	low-head safety injection
LiOH	lithium hydride
~~LMS~~	~~leak monitoring system~~
LOCA	loss-of-coolant accident
LOESW	loss of essential service water
LOF	left-out-force
LOFF	loss of feedwater flow
LOOP	loss of offsite power
LOP	loss of power
LPDS	large panel display (LPD) system
LPMS	loose parts monitoring system
LPSD	low-power and shutdown
LPT	low-pressure turbine
LPZ	low-population zone
LRB	last rotation blade
LRF	large release frequency
LRT	leakage rate testing
LS	lighting system
LSSS	limiting safety system settings
LTOP	low temperature overpressure protection
LWMS	liquid waste management system
LWR	light-water reactor
M signal	main control room isolation signal
M/D	motor-driven
M/G	motor generator
MAAP	modular accident analysis program
MACCS2	MELCOR accident Consequence Code system 2

ACRONYMS AND ABBREVIATIONS (Continued)

PPASS	process and post-accident sampling systems
PPS	preferred power supply
PRA	probabilistic risk assessment
PRDF	probabilistic risk assessment fundamental
PRDS	pressurizer and relief discharge system
PRS	pressure relief system
PRSV	pressurizer safety valve
PRT	pressurizer relief tank
PS	prestress
PS/B	power source building
PSB	power systems branch
PSF	performance shaping factor
PSFSV	power source fuel storage vault
PSI	preservice inspection
PSMS	protection and safety monitoring system
PSS	process and post-accident sampling system
PSWS	potable and sanitary water systems
PT	liquid penetrant examination method
PTFE	polytetrafluoroethylene
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
PWR	pressurized-water reactor
~~PWS~~	~~potable water system~~
PWSCC	prevention of primary water stress corrosion cracking
QA	quality assurance
QAP	quality assurance program
QAPD	quality assurance program document
QPTR	quadrant power tilt ratio
R/B	reactor building
RADTRAD	radionuclide transport, removal, and dose
RAI	request for additional information
RAP	reliability assurance program
RAT	reserve auxiliary transformer
RAW	risk achievement worth
RCA	radiological controlled area
RCC	rod control cluster
RCCA	rod cluster control assembly

ACRONYMS AND ABBREVIATIONS (Continued)

TMI	Three Mile Island
TPS	turbine protection system
TRANS	general transients
Tref	reference temperature
TRS	test response spectrum
TS	technical specification
TS	telecommunication system
TSC	technical support center
TSCVS	technical support center (TSC) HVAC system
TSIS	turbine supervisory instrument system
TT	thermal treatment
TVS	turbine building area ventilation system
U.S.	United States
UAT	unit auxiliary transformer
UB	upper bound
UCC	underclad cracking
UHS	ultimate heat sink
UHSRS	ultimate heat sink related structures
UL	Underwriters Laboratories
UPS	uninterruptible power supply
URD	Utility Requirement Document
US, U.S.	United States
USA	United States of America
USM	uniform support motion
UT	ultrasonic examination method
UTS	ultimate tensile strength
UV/IR	ultraviolet/infrared
V&V	verification and validation
VA	vital area
VAC	volts alternating current
VAS	auxiliary building ventilation system
VCS	containment ventilation system
VCT	volume control tank
~~VDS~~	~~vent drain system~~
VDU	visual display unit
VE	vital equipment
VFTP	ventilation filter testing program

(2) The fission products released from the sintered uranium dioxide pellet are sealed in the fuel cladding.

(3) Even if the fuel cladding is damaged, the leaked fission products are retained in the RCS.

(4) In case fission products are released due to failure of the RCS or some other mechanism, they are retained by the reactor containment composed of the containment vessel, the annulus portion and other elements of the containment.

Radioactive Waste Management Facilities are installed to treat and manage the radioactive waste produced as a result of plant operation, in order to keep the concentration and the quantity of radioactive substances released to the surrounding environment as low as reasonably acceptable.

1.2.1.2.6 Safety Protection System Design Criteria

The reactor trip system (RTS) and the engineered safety features system are composed of the reactor protection system (RPS), the engineered safety features actuation system (ESFAS), the safety logic system and the safety grade human system interface system, are designed to have redundancy and independency so as to actuate when necessary, and also are so designed that their protective functions are not prevented by a single failure. In addition, these systems are designed to fail to a safe state for all credible failures, such as loss of power and so forth.

1.2.1.2.7 Reactor Shutdown System Design Criteria

Two independent reactivity control systems of different design are provided. One of the systems uses control rods and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, (including anticipated operational occurrences), and with appropriate margin for malfunctions such as stuck rods, specified fuel design limits are not exceeded. The second reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The control rod drive system (CRDS) and the chemical and volume control system (CVCS) of the US-APWR are provided so that the core can be made subcritical during normal operation and be maintained subcritical. They are designed on the basis of the following principles:

(1) The CRDS is designed so that the core can be made hot subcritical from full power operation without exceeding acceptable fuel design limits.

(2) Proper operation of the CVCS prior to transients is assumed as an initial condition to evaluate the transients

Sodium tetraborate decahydrate (NaTB) contained in NaTB baskets provides adjustment of the pH of the water in the containment following an accident.Twenty three NaTB baskets containing NaTB as a buffer agent are located inside three NaTB basket containers at an elevation that is below the lowest spray ring. NaTB in baskets is dissolved in spray water in the containers. The solution containing NaTB is discharged from each container to the RWSP through NaTB solution transfer pipe.

Main Control Room (MCR) HVAC System - The MCR HVAC system is designed to provide proper environmental conditions for the control room envelope (CRE) during normal operation and accident conditions. The MCR HVAC system is also designed to protect operators against a postulated release of radioactive material and toxic gases.

The MCR HVAC system design is based on the followings:

- MCR HVAC system maintains proper environmental conditions for CRE in the normal operation mode and emergency modes.
- MCR HVAC system has two emergency modes; pressurization mode and isolation mode.
- Pressurization mode protects the MCR operators and staff within the CRE during the accident conditions.
- Isolation mode protects the MCR operators and staff within the CRE from external toxic gas or smoke.
- MCR HVAC system is powered from Class 1E busses so the safety functions can be maintained during a LOOP.

Annulus Emergency Exhaust System (AEES) - The AEES is designed for fission product removal and retention by filtering the air it exhausts from the annulus and safeguard component area following accidents.

The AEES satisfies the following design requirements:

- The AEES establishes and maintains a negative pressure in the annulus and safeguards component area relative to adjacent areas.
- The AEES removes and retains fission products by High-Efficiency Particulate Air (HEPA) filters and discharges exhaust air through the vent stack.
- The AEES is powered from Class 1E busses so its safety functions can be maintained during a LOOP.
- The AEES is automatically initiated by ECCS actuation signal.

1.2.1.5.4.2 Spent Fuel Pit Cooling and Purification System

- Communications Independence: Data communication between safety divisions or between safety and non-safety divisions does not inhibit the performance of the safety function.

The PSMS satisfies the following design requirements:

- PSMS consists of four train redundant ~~reactor protection system (~~RPS~~)~~, engineered safety features actuation system (ESFAS), safety logic system (SLS) and safety grade human-system interface system (HSIS), and conventional switches for system level manual actuation.

- Once initiated, either automatically or manually, protective functions proceed to completion. In addition, system level signals cannot be manually reset until the plant condition is restored to a pre-determined setpoint.

- The quality of PSMS components and modules and the quality of the PSMS design process is controlled by a program that meets the requirements of ASME NQA-1-1994.

- The PSMS is qualified for worst-case environmental and seismic requirements for the place of its installation. The PSMS qualification envelopes the environmental and seismic boundary conditions.

- Each train of the PSMS is independent from each other and from non-safety systems, including the PCMS. Electrical independence is maintained through qualified isolation devices, including fiber optic data communications cables. Functional independence between controllers is maintained through communication processors that are separate from function processors, and through (1) logic that assures prioritization of safety functions over non-safety functions and (2) logic that does not rely on signals from outside its own train to perform the safety function within the train.

- Testing from the sensor inputs of the PSMS through to the actuated equipment and HSI is accomplished through a series of overlapping sequential tests and calibrations. The majority of the tests are conducted automatically through self-diagnostics. Most remaining manual tests may be performed with the plant at full power.

- PAM Type A, B and C variables have redundant instrumentation and are displayed on at least two redundant safety-related visual display units .

- The PSMS system level bypassed or inoperable status indication is provided. These indications are displayed as the spatially dedicated continuously visible (SDCV) information on large display panel in the MCR.

- Software life cycle process is controlled using the software program manual to improve the functional reliability and design quality of software.

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 10 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(25)	*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS-SASSI ~~SSI Version 2.2~~ (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design. After the SASSI analysis is first performed for a specific unit, subsequent COLAs for other units may be able to forego SASSI analyses if the FIRS and GMRS derived for those subsequent units are much smaller than the US-APWR standard plant CSDRS, and if the subsequent unit can also provide justification through comparison of site-specific geological and seismological characteristics.*
COL 3.7(26)	*SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. Consideration of structure-to-structure interaction is discussed in Subsection 3.7.2.8. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:* • *Seismic category I ESWPT* • *Seismic category I PSFSV* • *Seismic category I UHSRS*
COL 3.7(27)	*It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.*
COL 3.7(28)	*The overall basemat dimensions, basemat embedment depths, and maximum height of the US-APWR R/B, PCCV, and containment internal structure on their common basemat are given in Table 3.7.1-3 and as updated by the COL Applicant to include site-specific seismic category I structures.*
COL 3.7(29)	*Table 3.7.2-1, as updated by the COL Applicant to include site-specific seismic category I structures, presents a summary of dynamic analysis and combination techniques including types of models and computer programs used, seismic analysis methods, and method of combination for the three directional components for the seismic analysis of the US-APWR standard plant seismic category I buildings and structures.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 13 of 44)

COL ITEM NO.	COL ITEM
COL 3.8(24)	*Other non-standard seismic category I buildings and structures of the US-APWR are designed by the COL Applicant based on site-specific subgrade conditions.*
COL 3.8(25)	*The site-specific COL are to assure the design criteria listed in Chapter 2, Table 2.0-1, is met or exceeded.*
COL 3.8(26)	*Subsidence and differential displacement may therefore be reduced to less than 2 in. if justified by the COL Applicant based on site specific soil properties.*
COL 3.8(27)	*The COL Applicant is to specify normal operating thermal loads for site-specific structures, as applicable.*
COL 3.8(28)	*The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures.*
COL 3.8(29)	*The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.*
COL 3.8(30)	*When a coefficient of friction of 0.7 is used in calculating sliding resistance F_s, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required.*
COL 3.9(1)	*The COL Applicant is to assure snubber functionality in harsh service conditions, including snubber materials (e.g., lubricants, hydraulic fluids, seals).*
COL 3.9(2)	*The first COL Applicant is to complete the vibration assessment program, including the vibration test results, consistent with guidance of RG 1.20. Subsequent COL Applicant need only provide information in accordance with the applicable portion of position C.3 of RG 1.20 for Non-Prototype internals.*
COL 3.9(3)	*Deleted*
COL 3.9(4)	*Deleted*
COL 3.9(5)	*Deleted*

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 1~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 1970)	Not applicable. SIP and CS/RHRP are designed so that adequate NPSH are provided to system pumps in accordance with Regulatory Guide 1.82 Rev.3.	N/A
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	Not applicable. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" is applied instead of Regulatory Guide 1.4.	N/A
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 1971)	Conformance with no exceptions identified.	8.1.5.3
1.7	Control of Combustible Gas Concentrations in Containment ~~Following a Loss-of-Coolant Accident~~ (Rev. 3, March 2007)	Conformance with no exceptions identified.	6.2.5.1, 19.2
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Revision 3, May 2000)	Not applicable. RG applies to a site-specific operational program.	12.1.1.3.1, 12.1.4
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants (Rev. 4, March 2007)	Conformance with no exceptions identified. US-APWR has no diesel generators, but will use gas turbine generators for emergency power in the standard design.	8.1.5.3, 14.2.12 (Note: MHI has generated a position on the use of gas turbine generators for emergency power that meets the intent of RG)
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 1971)	Conformance with exceptions. Isolation valve is not adopted to instrument lines for containment pressure.	6.2.4.1
1.12	Nuclear Power Plant Instrumentation for Earthquakes (Rev. 2, March 1997)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	3.7.4
1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, March 2007)	Conformance with no exceptions identified.	9.1.1, 9.1.2, 9.1.3, 9.1.4
1.14	Reactor Coolant Pump Flywheel Integrity (for Comment) (Rev. 1, August 1975)	Conformance with no exceptions identified.	5.4.1.1

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 2~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.16	Reporting of Operating Information – Appendix A Technical Specifications (Rev. 4, August 1975)	Withdrawn.~~Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.~~	N/A~~Chapter 16, 14.2.6, 14.2.7~~
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	Conformance with exceptions. The measurement at startup test for SG's internals is not planned.	3.9.2.3, 3.9.2.4, 3.9.2.6, 5.4.2.1.2.10, 14.2,
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	Conformance with exceptions. To be conformed by COL Applicant with site-specific information.	3.1.6, 9.3.2, 11.5.1, 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 1972)	Conformance with no exceptions identified.	7.1.3.11, 7.1.3.14, 8.1.5.3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Rev. 1, March 2007)	Not applicable. To be conformed by COL Applicant with site-specific characterization information.	N/A
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 1972)	Conformance with exceptions. To be conformed by COL Applicant with site-specific characterization information.	11.3.3
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 1972)	Not applicable. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" is applied instead of Regulatory Guide 1.25.	N/A
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	Conformance with no exceptions identified.	3.2.2, 5.2.1.1, 5.2.2.1, 5.2.4.1
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	Conformance with exceptions. US-APWR is designed in accordance with the functional requirements for a UHS as described in this RG, however design of the UHS is site-specific and will be the responsibility of the COL Applicant.	9.2.1.3, 9.2.5
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	Conformance with no exceptions identified.	14.2.7, 17.5

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 4~~8~~1 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.40	Qualification ~~Tests of~~ of Continuous-Duty Safety-Related Motors ~~Installed Inside the Containment of Water-Cooled~~ for Nuclear Power Plants ~~(~~(Rev. 1. February 2010~~0, March 1973~~)	Conformance with no exceptions identified.~~Not applicable.~~ ~~US-APWR has no Class 1 continuous-duty motors in the containment.~~	N/A
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (Rev. 0, March 1973)	Conformance with no exceptions identified.	8.1.5.3, 14.2.7
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.2.3.3.2, 5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	3.6.3.3.4, 5.2.3.4.1, 5.2.3.4.2, 6.1.1
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage ~~Reactor Coolant Pressure Boundary Leakage Detection Systems~~ (Rev. 1, May 2008)	Conformance with no exceptions Identified.	5.2.5, 11.5
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	Conformance with no exceptions identified.	8.1.5.3, table 8.1-1, 18.7.3.2, table 18.7-1
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	This RG has been withdrawn by NRC.	N/A
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	Conformance with ~~no~~ exceptions ~~identified.~~. The post-welded baking may be used in lieu of maintaining preheat until post weld heat treatment is performed.	5.3.1.2, 5.3.1.4, 5.2.3.3.2, 6.1.1
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	Conformance with no exceptions identified.	6.4.2, 6.4.6, Table 6.4-2, 6.5.1, Table 6.5-3, 9.4.1, 9.4.5, 12.3.3, 14.2.7
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	Conformance with no exceptions identified.	7.1.3. 2, 7.1.3.3, 8.1.5.3
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	Conformance with exceptions. Programmatic/operational and site-specific aspects are not applicable to US-APWR design certification. ASTM standard revision levels may differ from RG 1.54 as specifically referenced in the "Corresponding Chapter/Section/Subsection"	6.1.2 11.2 11.4
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	Not applicable. US-APWR has a concrete containment.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 6~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	Not applicable. US-APWR design does not use Spray Pond. The spray water of US-APWR is supplied from RWSP in containment.	N/A
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	Conformance with no exceptions identified.	8.1.5.3
1.75	Physical Independence of Electric Systems (Rev. 3, February 2005)	Conformance with no exceptions identified.	7.1.3, 8.1.5.3
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, table 2.0-1, 3.3.2, 3.5.1.4
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	Conformance with no exceptions identified. Note: The newer criteria for fuel cladding failure, core coolability and fission product inventory contained in SRP, Section 4.2, Appendix B will be used in conjunction with the requirements of RG 1.77 for the US-APWR Rod Ejection analysis.	15.4.8.2, 16.0
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	Conformance with exceptions. Full conformance by COL Applicant with site-specific consequence data.	6.4.4, 9.4.1
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	Conformance with no exceptions identified.	14.2.7
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Rev. 1, January 1975)	Not applicable. DCD describes a single reference plant design; RG applies to a site-specific multi-unit situation.	N/A
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Rev. 3, November 2003)	Conformance with exceptions. Full conformance by COL Applicant with site-specific conditions.	6.2.2.1
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	Withdrawn.~~Not applicable.~~ ~~This RG is considered for withdrawal by NRC. Current NRC requirements for this area are shown in steam generator program guidelines (NEI 97-06 Rev.2) and Technical Specification task Force TSTF-449 Rev.4.~~	N/A~~5.4.2.2~~
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III (Rev. 33, August 2005)	Conformance with no exceptions identified.	3.12.2.2

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 9~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, February 2007)	Conformance with exceptions. Criterion 5:OBE seismic evaluation is not required in US-APWR.	3.9.3.4, 3.12.6.1
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	Conformance with no exceptions identified.	2.4
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	Conformance with no exceptions identified	4.2, 4.4
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, March 1978)	Not applicable. RG applies to a site-specific operational program.	N/A
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with no exceptions identified. ~~The hydrogen concentration limit required in RG 1.189 is appropriate for the fire protection scenario, over the RG 1.128.~~	8.1.5.3, 14.2.7
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with exceptions. Design certification applicability is to assure design features accommodate functions described in RG; full conformance in terms of program and activities will be the responsibility of the COL Applicant.	8.1.5.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 2, March 2007)	Conformance with no exceptions identified.	3.9.3.4, 3.12.6.1
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	Withdrawn.~~Conformance with no exceptions identified.~~	N/A~~8.1.5.3~~
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	Not applicable. RG applies to site-specific operational program.	N/A
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	Conformance with exceptions. C.3.a: Section 13.5 defines the responsibility for development of administrative and operating procedures. C.6: The COL applicant has the responsibility of this requirement.	4.4.6.3
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants (Rev. 3, March 1998))	Not applicable. RG applies to a site-specific operational program.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 10~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	Withdrawn.~~Conformance with exception. Site-specific aspect is not applicable to US-APWR design certification.~~	N/A~~2.4~~
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments (Rev. 3, March 2007)	Conformance with no exceptions identified.	3.8.1.2, 14.2.7
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	Conformance with no exceptions identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	8.1.5.3, 9.5.4 (Note: MHI has generated a position on the use of gas turbine generators for emergency power that meets the intent of RG.)
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (Revision 2, December 2003)	Not applicable. RG applies to site-specific operational program.	N/A
1.139	Guidance for Residual Heat Removal (for Comment (Rev. 0, May 1978) (Note: Cold shutdown requirements as related to environmental qualification of equipment)	Withdrawn.~~Conformance with exceptions. Criterion 7 applies to a site-specific operational program.~~	N/A~~5.4.7, 6.3.1.3, 7.4.1~~
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, June 2001)	Conformance with no exceptions identified.	9.4.3, 9.4.6, 12.3.3, 14.2.7
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	Conformance with no exceptions identified.	6.2.4
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) (Rev. 2, November 2001)	Conformance with no exceptions identified.	3.5.3, 3.8.3, 3.8.4, 3.8.5
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	Conformance with no exceptions identified.	3.2.2, 11.2 11.3, 11.4
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	Not applicable. Full conformance by COL Applicant with site-specific dispersion data.	N/A
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (Rev. 14, August 2005)	Conformance with no exceptions identified.	5.2.1.2, 5.2.4.1, 5.2.4.2, 6.6.3
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)	Withdrawn.~~Conformance with no exceptions identified.~~	N/A~~3.9.6, 3.10.2~~

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 14~~1~~ of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	Not applicable. RG applies to a site-specific operational program that occurs during plant decommissioning and permanent shutdown.	N/A
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 0, June 2003)	Conformance with exceptions. RG is referenced specifically for applicability of code case OMN-13 requirements for snubber inspection.	3.9.3.4
1.193	ASME Code Cases Not Approved for Use (Rev. 1, August 2005)	Conformance with no exceptions identified.~~Not applicable.~~ ~~US-APWR design does not incorporate any of the identified ASME code cases.~~	N/A
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	Not applicable. Full conformance by COL Applicant with site-specific dispersion data	N/A
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	Not applicable. Due to use of alternative source term, the guidance of RG 1.183 is applied instead of RG 1.195.	N/A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	Conformance with no exceptions identified.	6.4
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	Conformance with no exceptions identified.	6.4.5
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (Rev. 0, November 2003)	Not applicable. RG applies to a site-specific analysis.	N/A
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	Conformance with no exceptions identified.	3.8.4, 3.9.3.4, 3.12.6.4
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 1, January 2007)	Conformance with no exceptions identified.	19.1
1.201	Guidelines for Categorizing SSCs in Nuclear Power Plants According to Their Safety Significance (Rev. 1, May 2006)	Conformance with exceptions. Note: No.3 and No.9 are the NRC position to meet the requirement of 10CFR50.69, but design of US-APWR is in accordance with the standard method described in RG 1.206, section C.I.3.	17.4, 19.1.7

Table 1.9.1-2 US-APWR Conformance with Division 4 Regulatory Guides (sheet 1 of 2)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
4.1	Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants (Rev. 1, April 1975)	Not applicable. RG applies to a site-specific environmental monitoring activity.	N/A
4.2	Preparation of Environmental Reports for Nuclear Power Stations (Rev. 2, July 1976)	Not applicable. RG applies to a site-specific environmental evaluation.	N/A
4.2S1	Supplement 1 to Regulatory Guide 4.2, Preparation of Supplemental Environmental Reports for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 0, September 2000)	Not applicable. RG applies to license renewals.	N/A
4.4	Reporting Procedure for Mathematical Models Selected To Predict Heated Effluent Dispersion in Natural Water Bodies (Rev. 0, May 1974)	Not applicable. RG applies to site-specific environmental activity of modeling temperature impact of plant discharge on aquatic systems.	N/A
4.5	Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil (Rev. 0, May 1974)	Withdrawn.~~Not applicable.~~ ~~RG applies to a site-specific environmental monitoring activity.~~	N/A
4.6	Measurements of Radionuclides in the Environment-- Strontium-89 and Strontium-90 Analyses (Rev. 0, May 1974)	Withdrawn.~~Not applicable.~~ ~~RG applies to a site-specific environmental monitoring activity.~~	N/A
4.7	General Site Suitability Criteria for Nuclear Power Stations (Revision 2, April 1998)	Not applicable. RG applies to a site-specific evaluation	N/A
4.8	Environmental Technical Specifications for Nuclear Power Plants (Rev. 0, December 1975)	Withdrawn.~~Not applicable.~~ ~~RG applies to a site-specific operational controls resulting from environmental characterization and commitments.~~	N/A
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities (Rev. 1, October 1975)	Not applicable. RG applies to uranium enrichment facilities.	N/A
4.11	Terrestrial Environmental Studies for Nuclear Power Stations (Rev. 1, August 1977)	Not applicable. RG applies to a site-specific environmental evaluation.	N/A
4.13	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications (Rev. 1, July 1977)	Not applicable. RG applies to a site-specific dosimetry product.	N/A
4.14	Radiological Effluent and Environmental Monitoring at Uranium Mills (Rev. 1, April 1980)	Not applicable. RG applies to uranium mills.	N/A

Table 1.9.1-4 US-APWR Conformance with Division 8 Regulatory Guides (sheet 1 of 4)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
8.2	Guide for Administrative Practices in Radiation Monitoring (Rev. 0, February 1973)	Conformance with exceptions. To be conformed in COL.	12.1.4, 12.3.4
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters (Rev. 0, February 1973)	Conformance with exceptions. To be conformed in COL.	12.1.4
8.5	Criticality and Other Interior Evacuation Signals (Rev. 1, March 1981)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A
8.6	Standard Test Procedure for Geiger-Muller Counters (Rev. 0, May 1973)	Withdrawn.~~Conformance with exceptions.~~ ~~To be conformed in COL.~~	N/A~~12.1.4~~
8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data (Rev. 2, November 2005)	Conformance with exceptions. To be conformed in COL.	12.1.4
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable (Rev. 3, June 1978)	Conformance with exceptions. All design issues are addressed; site-specific policy considerations are outside scope of design certification.	3.7.4.2, 9.3.2, 11.3.1, 11.4.1, 11.4.2, , 12.1.1.3, 12.1.2, 12.1.4, 12.2.1.1.10, 12.3.1, 12.3.2.1, 12.3.2.2, 12.3.3.3, 12.3.4,
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Rev. 1, July 1993)	Not applicable. RG refers to site-specific procedures and/or equipment that are outside the reference US-APWR design.	N/A (RG is mentioned, however, in 12.1.4)
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Rev. 1-R, May 1977)	Conformance with exceptions. Programmatic/operational aspect is not applicable to US-APWR design certification.	12.1.1.3, 12.1.4, 12.2.1.1.10
8.11	Applications of Bioassay for Uranium (Rev. 0, June 1974)	Not applicable. RG applies to bioassay for uranium.	N/A

Table 1.9.2-6 US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features (sheet 25~~25~~ of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
6.5.4 Ice Condenser as a Fission Product Cleanup System	The acceptance criteria for the fission product cleanup function of ice condenser system are based on the relevant requirements of the following regulations: A. General Design Criterion 41 B. General Design Criterion 42 C. General Design Criterion 43	Not applicable. US-APWR does not have the ice condenser containments.	N/A
6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System	This SRP applies to boiling water reactors and is not applicable to the US-APWR.	Not applicable. Requirements apply only to BWRs.	N/A
6.6 Inservice Inspection and Testing of Class 2 and 3 Components	1. Components Subject to Inspection 2. Accessibility 3. Examination Categories and Methods 4. Inspection Intervals 5. Evaluation of Examination Results 6. System Pressure Tests 7. Augmented ISI to Protect Against Postulated Piping Failures 8. Code Exemptions 9. Relief Requests 10. Code Cases 11. Operational Programs	Conformance with exceptions. Criteria ~~8,9,10 and~~ 11:~~There are discussed in inservice inspection program prepared for~~This is a COL application responsibility.	~~6.5.2~~6.6
6.7 Main Steam Isolation Valve Leakage Control System (BWR)	This SRP applies to boiling water reactors and is not applicable to the US-APWR	Not applicable. Requirements apply only to BWRs.	N/A
Branch Technical Position 6-1: pH For Emergency Coolant Water for Pressurized Water Reactors	1. Minimum pH should be 7.0. 2. For the spray water recirculated from the containment sump, the higher the pH in the 7.0 to 9.5 range, the greater the assurance that no stress corrosion cracking will occur. See SRP Section 6.5.2 for additional water chemistry requirements related to fission product removal. 3. If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.	Conformance with no exceptions identified.	6.1.1.2, 6.3.1.3

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 15 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-13: Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Supporting Analysis (Additional text follows on requirements) • Traceability of the Installed Reference RTD to Laboratory Calibration Data (Additional text follows on requirements) • Acceptable Methods for In-Situ Testing (Additional text follows on requirements) • Response Time Testing (Additional text follows on requirements) • Control/Protection Interaction and Common-Cause Failure During In-Situ Testing *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7
Branch Technical Position 7-14: Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems	B.3.1 Acceptance Criteria for Planning (Additional text follows on requirements) B.3.2 Acceptance Criteria for Implementation *(Additional text follows on requirements)* B.3.3 Acceptance Criteria for Design Outputs *(Additional text follows on requirements)*	Conformance with no exceptions identified.	7.1.3.17, 7.9.2.2, 7.9.2.6
Branch Technical Position 7-16 (Withdrawn) Guidance on Level of Detail Required for Design Certification Applications Under 10CFRPart 52	This BTP has been withdrawn.	Not applicable. SRP has been withdrawn by NRC.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (sheet 10 of 30)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.5 Ultimate Heat Sink (continued)	6. 10CFR52.47(b) (1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. 7. 10CFR52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.2.6 Condensate Storage Facilities	1. Protection Against Natural Phenomena. Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding. 2. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CSF in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Not applicable. ~~US-APWR design does not have a condensate storage system.~~Condensate Storage Facilities has no safety related functions. US-APWR is not multiple – unit.	~~N/A~~9.2.6

Chapter 3

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	**Description of Change**
3-xv	APPENDICES Appendix 3H	Change: “Model Properties and Seismic Analysis Results for Lump Mass Stick Models of R/B-PCCV-Containment Internal Structure on a Common Basemat, PS/Bs on Individual Basemats” to “Model Properties for Lump Mass Stick Models of R/B-PCCV-Containment Internal Structure on a Common Basemat” Reason: Editorial correction of title
3-xvi	Tables Table 3.7.2-3	Change: “Seismic SSI Analysis Cases” to “Deleted” Reason: Editorial Correction
3-xvii	Tables Table 3.9-3	Change: “Minimum Design Loading Combinations for ASME Code, Section III, Class 1, 2, 3 and CS Systems and Components” to “Design Loading Combinations for ASME Code, Section III, Class 1, 2, 3 and CS Systems and Components” Reason: Editorial Correction
3-xvii	Tables Table 3.9-4	Change: “Minimum Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components” to “Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components” Reason: Editorial Correction
3-xix	Figures Figure 3.7.2-11	Change: “Development of Enveloped Design ISRS” to “Deleted” Reason: Editorial Correction
3-xix	Figures Figure 3.7.2-12	Change: “Example Design ISRS” to “Deleted” Reason: Editorial Correction
3-xxvi	ACRONYMS AND ABBREVIATIONS	Deleted ELS (emergency letdown system) Reason: Editorial correction
3-xxvi	ACRONYMS AND ABBREVIATIONS	Added FSS (fire protection water supply system) Reason: Editorial correction
3-xxvii	ACRONYMS AND ABBREVIATIONS	Change: “rod control cluster assembly” to “rod cluster control assembly” Reason: Editorial correction to modify the abbreviation of RCCA (rod cluster control assembly)

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.1-22	Subsection 3.1.4.4.1 3rd Paragraph 1st Sentence	Change: "The emergency letdown system (ELS) consists of two…" to "The emergency letdown system consists of two…" Reason: Deleted the acronym of emergency letdown system.
3.1-24	Subsection 3.1.4.6.1 1st Paragraph	Change: "The ECCS of the US-APWR includes the accumulator system, HHIS and ELS." to "The ECCS of the US-APWR includes the accumulator system, HHIS and emergency letdown system." Reason: Deleted the acronym of emergency letdown system.
3.1-24	Subsection 3.1.4.6.1 3rd Paragraph 1st Sentence	Change: "The Discussion section of GDC 33 describes the ELS, the system's flow…" to "The Discussion section of GDC 33 describes the emergency letdown system, the system's flow…" Reason: Deleted the acronym of emergency letdown system.
3.2-19	Table 3.2-2 Sheet 3 6th Row	Change in the 6th Column: "N/A" to "4" Change in the 7th Column: "NS" to "II" Reason: Corrected the description of "Codes and Standards" and "Seismic Category".
3.2-20	Table 3.2-2 Sheet 4 5th Row 1st Column	Change: "Pressurizer spray valves RCS-PCV451A, B" to "Pressurizer spray valves RCS-PCV-061A,B" Reason: Corrected the valve number.
3.2-25	Table 3.2-2 Sheet 9 3rd Row	Add new 4th Row: "Letdown line piping and valves outside containment from CVS-VLV-102, VLV-103 (including the valves) to the volume control tank"; "4"; "R/B"; "D"; "N/A"; "4"; "II" Reason: Correct erroneously omitted information
3.2-26	Table 3.2-2	Change: "Reactor coolant pump seal water injection piping and valves excluding containment isolation

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 10 3rd Row	valves, piping between these valves, piping downstream of CVS-VLV-180A, B, C, D (including valves), and seal water injection filter line valves and piping between and excluding CVS-VLV-168 and CVS-VLV-173" to "Reactor coolant pump seal water injection piping and valves excluding containment isolation valves, piping between these valves, piping downstream of CVS-VLV-180A, B, C, D (excluding valves), and seal water injection filter line valves and piping between and including CVS-VLV-168 and CVS-VLV-173" Reason: Editorial correction for the correct use of including and excluding
3.2-29	Table 3.2-2 Sheet 13 Last Row 1st Column	Change: "CVS-FCV-133A, 129 and CVS-VLV-581" to "CVS-FCV-133A, 129, 128 and CVS-VLV-581" Reason: Provided consistency between Tier 1 and Tier 2.
3.2-32	Table 3.2-2 Sheet 16 8th Row 7th Column	Change: "NS" to "II" Reason: Editorial correction
3.2-33	Table 3.2-2 Sheet 17 11th Row 7th Column	Change: "II" to "NS" Reason: Editorial correction
3.2-34	Table 3.2-2 Sheet 18 5th Row 1st Column	Change: "EFS-VLV-109A,B" to "EFS-VLV-109A,B,C,D" Reason: Engineering design/analysis development.
3.2-34	Table 3.2-2 Sheet 18 Last Row 7th Column	Change: "NS" to "II" Reason: Editorial correction
3.2-35	Table 3.2-2	Change: "NS" to "II"

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 19 2nd-3rd Rows 7th Column	Reason: Editorial correction
3.2-35	Table 3.2-2 Sheet 19 7th Row	Change: Valve ID “NFS-SMV-512A,B,C,D…” to “FWS-SMV-512A,B,C,D…” Reason: Correct valve ID.
3.2-35	Table 3.2-2 Sheet 19 8th Row	Change: Valve ID “NFS-VLV-512A,B,C,D…” to “FWS-SMV-512A,B,C,D…” Reason: Correct valve ID.
3.2-37	Table 3.2-2 Sheet 21 3rd Row	Change: Valve ID “MSS-VLV-509A,B,C,D…” to “MSS-SRV-509A,B,C,D…” Reason: Correct valve ID.
3.2-39	Table 3.2-2 Sheet 23 2nd Row 1st Column	Change: “EFS-VLV-109A,B” to “EFS-VLV-109A,B,C,D” Reason: Engineering design/analysis development.
3.2-55	Table 3.2-2 Sheet 39 4th Row	Change: “RHS loop sampling piping and valves up to and including the valves PSS-MOV-052A,B” to “RHS loop sampling piping and valves up to and including the valves PSS-MOV-052A,B,C,D” Reason: Correct erroneously omitted information
3.2-55	Table 3.2-2 Sheet 39 10th Row	Add new 10th Row: “Containment vessel atmosphere sampling inlet, outlet valve PSS-MOV-301, 312”; “4”; “R/B”; “D”; “N/A”; “4”; “I” Reason: Correct erroneously omitted information
3.2-55	Table 3.2-2 Sheet 39 11th Row	Add to the end of the Systems and Components description: “(excluding PSS-MOV-301, 312)” Reason: Editorial correction
3.2-55	Table 3.2-2 Sheet 39	Add two rows for “PSS-MOV-301” and “PSS-MOV-312” Reason: Correct erroneously omitted information

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	12th and 13th Rows	
3.2-56	Table 3.2-2 Sheet 40 Item 26 1st Row	This portion is modified by designated as CDI: Change: "R/B, A/B, AC/B PS/B, T/B" to "R/B, A/B, AC/B, T/B" Reason: No sanitary water system is needed in PS/B.
3.2-63	Table 3.2-2 Sheet 47 6th Row	Change in the 2nd Column: "5" to "3" Change in the 4th Column: "N/A" to "C" Change in the 5th Column: "N/A" to "YES" Change in the 7th Column: "II" to "I" Reason: Revised the description (e.g., Equipment Class, Quality Group and Seismic Category etc.) for the duct heaters of Class 1E Electrical Room HVAC System due to changing of the safety classification for the duct heaters.
3.5-10	Subsection 3.5.1.3.2 3rd Paragraph 2nd Sentence	Change: "For favorably oriented T/Gs as outlined in the geometry Section 3.5.1.3, the product of P_2 and P_3 is conservatively estimated as 10^{-3} per year." to "For the geometry of Subsection 3.5.1.3 the product of P_2 and P_3 is estimated as 10^{-2} per year, which is a more conservative estimate than for a favorably oriented single unit and in conformance with the guidance in SRP Section 3.5.1.3. This conservative estimation provides the flexibility for the orientation of site-specific SSCs of concern based on the guidance of RG 1.117 (Reference 3.5-19) and RG 1.115 (Reference 3.5-6)." Reason: Clarify application of probability limits. [RAI 323-2071 Question 03.05.01.03-3] Included reference to RG 1.115 and RG 1.117 [RAI 598-4754 Question 10.02-4]
3.5-10	Subsection 3.5.1.3.2 3rd Paragraph Last Sentence	Change: "…equal or less than the acceptable limit of 1 $\times 10^{-5}$" to "…less than the acceptable limit of 1 $\times 10^{-5}$". Reason: Editorial correction.
3.6-13	Subsection	Change: "For ASME Code, Section III, Class 1 piping,

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	3.6.2.1.2.2 1st Paragraph 1st Bullet	where the stress range calculated by Eq. (10) in NB-3653 is less than 1.2 S(m)" to "For ASME Code, Section III, Class 1 piping, where the stress range calculated by Eq. (10) in NB-3653 is more than 1.2 S(m)" Reason: Editorial Correction [RAI 636-4732, Question 03.06.02-47]
3.6-15	Subsection 3.6.2.3 2nd Paragraph	Change: "The analytical methods used for the calculation of the jet thrust for the above described situations are based on SRP 3.6.2 (Reference 3.6-3) and ANSI/ANS 58.2-1988 (Reference 3.6-14)." to "The analytical methods used for the calculation of the jet thrust for the above described situations are based on SRP 3.6.2 (Reference 3.6-3) and MHI original methodologies (Reference 3.6-25) based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31." Reason: Updated references [RAI 636-4732, Question 03.06.02-41]
3.6-18	Subsection 3.6.2.4.1 2nd Paragraph	Add as 2nd Sentence: "The Jet impingement pressure essentially has non-uniform distributions, which varies with distance from the pipe break as shown in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31. However, the maximum pressure in the non-uniform distribution is conservatively used as a uniform distribution." Reason: Added text clarifies jet impingement pressure distribution and application. [RAI 636-4732, Question 03.06.02-42]
3.6-18	Subsection 3.6.2.4.1 3rd Paragraph 1st Sentence	Change: "The methods used to evaluate the jet effects resulting from the postulated breaks in high energy piping are described in Appendices C and D of ANSI/ANS 58.2 (Reference 3.6-14)." to "The MHI original methodologies (Reference 3.6-25) used to evaluate the jet effects resulting from the postulated breaks in high energy piping are based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31." Reason: Added references for MHI methods and measurements [RAI 636-4732, Question 03.06.02-41]
3.6-18	Subsection 3.6.2.4.1	Add new Subsection 3.6.2.4.1.1: "**3.6.2.4.1.1 Blast Wave Assessing Procedure**

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Computational Fluid Dynamics analysis confirms the generation of a blast wave from a steam pipe break. Potential effects are assessed on equipment within the US-APWR pressurizer compartment. Distance between the postulated pipe break locations and components is long enough to attenuate the effects. However, if layout in the pressurizer compartment is changed in the future, reassessment of the blast wave will be conducted. Blast wave is not considered to occur from a sub-cooled water pipe break. This is because velocity of the two-phase flow at break point is slower than the speed of sound in atmospheric environments. Therefore, the blast wave does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing a blast wave from a steam pipe break." Reason: Subsection added to provide generalized blast wave assessment procedure [RAI 636-4732, Question 03.06.02-40]
3.6-19	Subsection 3.6.2.4.1	Add new Subsection 3.6.2.4.1.2: "**3.6.2.4.1.2 Jet Pressure Oscillation Assessing Procedure** Jet pressure oscillation from a steam pipe break is unlikely to occur in the US-APWR due to its high compression ratio. The jet flow expansion and Mach Disk is large. This leads to a stable downstream after the Mach Disk. The flow is so stable that disturbance at the impingement wall does not reach back to the Mach Disk. When sub-cooled jet-flow impinges on the wall, pressure distributions on the wall are not of the concave type and a re-circulation vortex is not generated. It is because flow velocity at the jet boundary is lower than that of the core region. Therefore, jet pressure oscillation does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing jet pressure oscillation from a steam pipe break." Reason: Subsection added to provide generalized jet pressure oscillation procedure [RAI 636-4732 Question

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		03.06.02-44]
3.6-19	Subsection 3.6.2.4.1	Add new Subsection 3.6.2.4.1.3: "**3.6.2.4.1.3 Jet Reflection Assessing Procedure** When jet flow impinges on a perpendicular wall, impinged jet flow is redirected and runs along the surface of the wall. Zone of influence obtained by computational fluid dynamics is enveloped by the estimated zone of influence from MHI original methodologies (Reference 3.6-25). Inside the zone of influence, impingement pressure includes effects of pressure due to flow parallel to an impingement wall. Loads due to jet impingement reflection outside of the zone of influence are considered so small that it is not necessary to be considered. Therefore, jet reflection does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing jet reflection." Reason: Added subsection to provide generalized jet reflection assessing procedure [RAI 636-4732, Question 03.06.02-45]
3.6-23	Subsection 3.6.2.6	Add new subsection 3.6.2.6: "**3.6.2.6 Outline of Pipe Break Hazard Analysis Report(s)** The following information provides an outline of methodology for the pipe break hazard analysis that will be completed for high and moderate energy piping systems (including the non-safety class piping) identified in Table 2.3-1 for the closure of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) Tier 1, Table 2.3-2 related to the pipe break hazard analysis report: • Identification of pipe break locations in high energy piping[1] • Identification of leakage crack locations in high and moderate energy piping • Identification of SSCs that are safety-related or required for safe shutdown[2] • Evaluation of consequences of pipe whip and jet impingement

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		• Evaluation of consequences of spray wetting, flooding, environmental conditions • Design and location of protective barriers, restraints, and enclosures Notes 1. Table 3.6-2 provides the list of high energy lines for pipe break hazard analysis, including properties of internal and external fluids. 2. All the SSCs that are safety-related or required for safe shutdown in close proximity to the postulated pipe rupture will be identified." Reason: Added subsection to provide generalized outline of Pipe Break Hazards Analysis Report(s). [RAI 636-4732, Question 03.06.02-48]
3.6-36	Subsection 3.6.5 Reference 3.6-25 to Reference 3.6-31	Add new references: "3.6-25 MUAP-10017, Revision 1, US-APWR Methodology of Pipe Break Hazard Analysis, December, 2010. 3.6-26 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Study of Pipe Reaction Force and Jet Impingement Load at the Pipe Break, Trans. 5th Int. Conf. on SmiRT, F6/2, 1979. 3.6-27 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Studies on Transient Water-Steam Impinging Jet, Vol. 22 No. 5, pp. 403-409, Journal of Atomic Energy Society of Japan, 1980 (in Japanese). 3.6-28 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Studies on Steam Free Jet and Impinging Jet, Vol. 22 No. 9, pp. 634-640, Journal of Atomic Energy Society of Japan, 1980 (in Japanese). 3.6-29 Masuda, F., Nakatogawa, T., Kawanishi, K. and Isono, M., Experimental Study on an Impingement High-Pressure Steam Jet, Nuclear Engineering and Design 67-2, pgs 273-285, 1982. 3.6-30 Masuda, F., Nakatogawa, T., Kawanishi, K. and Isono, M., Experimental Study on Jets Formed Under Discharges of High-Pressure Subcooled Water and Steam-Water Mixture, Trans. 7th Int. Conf. on SmiRT, F1/6, 1983.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		3.6-31 Isozaki, T. and Miyazono, S., Experimental Study of Jet Discharging Test Results under BWR and PWR Loss of Coolant Accident Conditions, Nuclear Engineering and Design 96, 1986." Reason: Added References for MHI methodologies and measurements [RAI 636-4732, Question 03.06.02-41]
3.6-36	Subsection3.6.5 Reference 3.6-32	Add new reference: "3.6-32 MUAP-10022, Revision 0, Evaluation on Jet Impingement Issues Associated with Postulated Pipe Rupture, January, 2011." Reason: Reference added consistent with corresponding subsection addition [RAI 636-4732, Question 03.06.02-40]
3.6-37	Table 3.6-1 22nd Row	Change: "Fire Service System" to "Fire Protection Water Supply System" Reason: Editorial Correction
3.6-38	Table 3.6-2 All 4 Sheets	Add Table 3.6-2. Reason: Provided list of high energy lines for pipe break hazard evaluation including properties of internal and external fluids. [RAI 636-4732, Question 03.06.02-43]
3.7-10	Subsection 3.7.1.3 2nd Paragraph Last Sentence	Change: "A minimum factor of safety of 2.5 is suggested for the ultimate bearing capacity versus the allowable dynamic bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions." to "A minimum factor of safety of 2.5 is suggested for the ultimate bearing capacity versus the allowable static bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions. A minimum factor of safety of 2 is suggested for the ultimate bearing capacity versus the allowable dynamic bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions." Reason: Editorial Correction
3.7-11	Subsection 3.7.1.3 6 Paragraph	Change: "The site-specific SSI analyses of the R/B–PCCV-containment internal structure on their common basemat uses the finite element (FE) analysis program

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	4th Sentence	SASSI Version 2.2 (Reference 3.7-17) that provides…" to "The site-specific SSI analyses of the R/B–PCCV-containment internal structure on their common basemat uses the finite element (FE) analysis program ACS SASSI (Reference 3.7-17) that provides…" Reason: Updated SASSI reference to be consistent with design bases version of SASSI documented in MUAP-10001 and -10006. [RAI 660-5134, Question 03.07.02-52]
3.7-11	Subsection 3.7.2 2nd Paragraph	Add as first sentence: "The seismic responses of the major seismic category I and seismic category II structures are required to be obtained from frequency domain time history analysis of seismic models considering a frequency-dependent SSI system, including a set of eight generic layered soil profiles representing a wide range of site conditions." Reason: Update to be consistent with current design and analysis approaches.
3.7-17	Subsection 3.7.2.3.2 1st Paragraph 3rd Sentence	Change: "The methodology initially used to develop the stick models and the stick model properties is presented in Technical Report MUAP-08005 (Reference 3.7-18), and in the following Subsections 3.7.2.3.4 through 3.7.2.3.9." to "The methodology initially used to develop the stick models and the stick model properties is presented in Technical Report MUAP-10001 (Reference 3.7-47), and in the following Subsections 3.7.2.3.4 through 3.7.2.3.9." Reason: Inclusion of lumped mass stick model in MUAP-10001.
3.7-19	Subsection 3.7.2.3.4 2nd Paragraph Last Sentence	Change: "This is the approach used for including the RCL seismic subsystem in the coupled RCL-R/B-PCCV-containment internal structure lumped mass stick model discussed in Technical Report MUAP-08005 (Reference 3.7-18)." to "This is the approach used for including the RCL seismic subsystem in the coupled RCL-R/B-PCCV-containment internal structure lumped mass stick model discussed in Technical Report MUAP-10001 (Reference 3.7-47)." Reason: Inclusion of lumped mass stick model in MUAP-10001.
3.7-28	Subsection 3.7.2.4	Change: "PS/Bs" to "PS/B model"

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	1st Paragraph Last sentence	Reason: Editorial correction.
3.7-29	Subsection 3.7.2.4.1 1st Paragraph 1st Sentence	Change: “The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure utilizing the program ACS-SASSI SSI Version 2.2 (Reference 3.7-17) which contains time history input incoherence function capability.” to “The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS SASSI (Reference 3.7-17) which contains time history input incoherence function capability.” Reason: Fulfills commitment to provide site-specific SSI analysis for PS/B. [RAI 495-3980, Question 3.7.2-4] and Updated SASSI reference to be consistent with design bases version of SASSI documented in MUAP-10001 and -10006. [RAI 660-5134, Question 03.07.02-52]
3.7-32	Subsection 3.7.2.5 1st Paragraph	Replace 1st Paragraph in its entirety with the following: “ISRS for the PS/Bs and RCL-R/B-PCCV-containment internal structure are developed from the results of the site-independent seismic analyses of the seismic models described in Subsection 3.7.2.3 by applying methods described in Subsection 3.7.2.1, and capturing SSI effects as described in Subsection 3.7.2.4, using generic soil profiles described in Subsection 3.7.1.3. The statistically independent time histories developed from the CSDRS as described in Subsection 3.7.1.1 serve as input control motion in the analysis. Note that the dynamic properties of the stick model portions of the R/B complex seismic model presented in Technical Report MUAP-10001 (Reference 3.7-47) are modified to account for the effects of cracking for accuracy in the seismic design and development of the ISRS. The ISRS are derived from the calculated responses at locations and elevations where major seismic category I and II SSCs are located.” Reason: Update ISRS based on updated seismic analysis methodologies described in technical reports MUAP-10001 and 10006. [RAI 660-5134, Questions 03.07.02-63 and -65]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.7-32	Subsection 3.7.2.5 2nd Paragraph Last Sentence	Change: “The local analyses of floor slab systems with respect to out-of-plane flexibility and effects on the ISRS are addressed as part of a later Technical Report.” to “The local analyses of floor slab systems with respect to out-of-plane flexibility and effects on the ISRS are addressed in Technical Reports MUAP-10001 and MUAP-10006 (References 3.7-47 and 3.7-48).” Reason: Update ISRS based on updated seismic analysis methodologies described in technical reports MUAP-10001 and 10006. [RAI 660-5134, Questions 03.07.02-63 and -65]
3.7-33	Subsection 3.7.2.5 4th Paragraph 1st Sentence	Change: “…IEEE Std 344-1987 (Reference 3.7-25),…” to “…IEEE Std 344-2004 (Reference 3.7-13),…” Reason: RG 1.100, Revision 3 Update
3.7-33	Subsection 3.7.2.5 5th Paragraph 2nd to last Sentences	Change: “The ISRS envelope the spectra obtained from the site-independent analyses for all four of the different generic subgrade conditions. Figure 3.7.2-11 outlines the development of the enveloped design ISRS, for which Figure 3.7.2-12 provides an example of a design ISRS. Design ISRS for R/B-PCCV-containment internal structure are provided in Appendix 3I. The process for developing enveloped ISRS is as follows:” to “The ISRS envelope the spectra obtained from the site-independent analyses for all generic subgrade conditions described in Subsection 3.7.1.3. ISRS developed from the site-independent seismic analyses of the R/B complex and PS/Bs are used for design. ISRS developed at 5% critical damping, which are presented in Technical Report MUAP-10006 (Reference 3.7-48) are used to validate the standard plant ISRS by comparison to site-specific ISRS that are also developed at 5% critical damping. The process for developing enveloped ISRS is described in detail in Section 3.5 of Technical Report MUAP-10006 and is summarized as follows:” Reason: Update ISRS based on updated seismic analysis methodologies described in technical reports MUAP-10001 and 10006. [RAI 660-5134, Questions 03.07.02-63 and -65]
3.7-50	Subsection 3.7.3.1.7.3	Change: “…IEEE Std 344-1987 (Reference 3.7-25)…” to “…IEEE Std 344-2004 (Reference 3.7-13)…”

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Last Paragraph Last Sentence	Reason: RG 1.100, Revision 3 Update
3.7-53	Subsection 3.7.4.1 1st Paragraph 2nd Sentence	Change: “The seismic design of US-APWR standard plant is based on site-independent seismic response analysis of basemat resting on the surface of elastic half-space that is subjected to a control motion.” to “The seismic design of US-APWR standard plant is based on site-independent seismic response analysis of basemats resting on generic supporting media that are subjected to the CSDRS input control motion.” Reason: Update the discussion to be consistent with the current design and analysis approaches
3.7-53	Subsection 3.7.4.1 1st Paragraph Last Sentence	Change: “FIRS, which are developed from site-specific GMRS, define the site-specific control design motion.” to “The FIRS, which are developed consistent with the site-specific GMRS define the site-specific control design motion.” Reason: Update the discussion to be consistent with the current design and analysis approaches
3.7-55	Subsection 3.7.4.2 6th Paragraph Last Sentence	Change: “…IEEE Std 344-1987 (Reference 3.7-25);…” to “…IEEE Std 344-2004 (Reference 3.7-13);…” Reason: RG 1.100, Revision 3 Update
3.7-60	Subsection 3.7.5 *COL3.7(25)* 1st Sentence	Change: “*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure utilizing the program ACS-SASSI SSI Version 2.2 (Reference 3.7-17) which contains time history input incoherence function capability.*” to “*The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS SASSI (Reference 3.7-17) which contains time history input incoherence function capability.*” Reason: Fulfills commitment to provide site-specific SSI analysis for PS/B. [RAI 495-3980, Question 3.7.2-4]
3.7-62	Subsection 3.7.6 Reference 3.7-13	Change: “<u>IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations</u>, IEEE Std 344-2004, Appendix D, Institute of Electrical and Electronic Engineers Power

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Engineering Society, New York, New York, June 2005." to "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005." Reason: RG 1.100, Revision 3 Update
3.7-62	Subsection 3.7.6 Reference 3.7-17	Change: "An Advanced Computational Software for 3D Dynamic Analysis Including Soil-Structure Interaction, ACS SASSI PREP User's Guide, Revision 2, for ACS SASSI, Version 2.2, Ghiocel Predictive Technologies, Inc. Pittsford, NY." to "An Advanced Computational Software for 3D Dynamic Analysis Including Soil-Structure Interaction, ACS Version 2.3.0, June 2009, and User Manual Revision 1, August 31, 2009, Ghiocel Predictive Technologies, Inc. Pittsford, NY.." Reason: Updated version of SASSI to be consistent with the design basis version of SASSI documented in MUAP-10001 and -10006.
3.7-62	Subsection 3.7.6 Reference 3.7-18	Change: "Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model, MUAP-08005 Rev. 0, April 2008" to "Deleted" Reason: MUAP-08005 has been superseded by MUAP-10001 and -10006
3.7-63	Subsection 3.7.6 Reference 3.7-25	Change: "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-1987, The Institute of Electrical and Electronics Engineers, Inc, New York, New York, 1987." to "Deleted" Reason: RG 1.100, Revision 3 Update
3.7-64	Subsection 3.7.6 References 3.7-47 and 3.7-48	Add 2 new references: "3.7-47 Seismic Design Bases of the US-APWR Standard Plant, MUAP-10001, Revision 2, Mitsubishi Heavy Industries, Ltd., January 2011. 3.7-48 Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant, MUAP-10006, Revision 1, Mitsubishi Heavy Industries, Ltd., January 2011."

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Reason: Updated analysis methodology
3.7-65	Table 3.7.1-1	Added Control Point E to the table. Reason: To be consistent with the design basis in MUAP-10001
3.7-66	Table 3.7.1-2	Added Control Point E to the table. Reason: To be consistent with the design basis in MUAP-10001
3.7-68	Table 3.7.1-4	Change the maximum Vertical value for 10-100Hz from “1.82” to “1.182” Reason: Editorial correction
3.7-69	Table 3.7.1-5 3rd Row 4th Column	Change: “7.14” to “7.145” Reason: Editorial correction
3.7-71	Table 3.7.2-1 3rd Row 2nd Column	Change: “Time History Analysis” to “Time History Analysis in Frequency Domain using sub-structuring technique” Reason: Editorial correction
3.7-71	Table 3.7.2-1 4th Row 4th Column	Change: “N/A(1)” to “N/A(2)” Reason: Editorial correction
3.7-72	Table 3.7.2-3	Change Title: “Seismic SSI Analysis Cases” to “Deleted” Delete Table in its entirety Reason: To be consistent with the design basis in MUAP-10001
3.7-75	Figure 3.7.1-1	Replace figure in its entirety Reason: To be consistent with the design basis in MUAP-10001
3.7-76	Figure 3.7.1-2	Replace figure in its entirety Reason: To be consistent with the design basis in MUAP-10001
3.7-94	Figure 3.7.2-11	Delete figure in its entirety

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Change title “Development of Enveloped Design ISRS” to “Deleted” Reason: To be consistent with the design basis in MUAP-10001
3.7-95	Figure 3.7.2-12	Delete figure in its entirety Change title “Example Design ISRS” to “Deleted” Reason: To be consistent with the design basis in MUAP-10001
3.8-5	Subsection 3.8.1.3.1 1st Bullet	Add as last 2 sentences: “The minimum prestress level including all losses after design life applied to the PCCV is 1.20 times the design pressure. The minimum prestress level including all losses after design life applied to the PCCV is 1.20 times the design pressure.” Reason: Add minimum prestress level [RAI 635-4954, Question 06.02.05-40]
3.8-7	Subsection 3.8.1.3.2 2nd-Last Paragraphs	Replace in its entirety with: “For the factored load design associated with the prestressed concrete wall: $D + P_g1 + [P_g2 \text{ or } P_g3]$ where D = Dead load P_g1 = Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction = 46.7 psia from Reference 3.8-55 P_g2 = Pressure resulting from uncontrolled hydrogen burning (if applicable) = 127 psia from Reference 3.8-55 P_g3 = Pressure resulting from post-accident inerting assuming carbon dioxide is the inerting agent (Not applicable to US-APWR) The factored load design of the US-APWR PCCV complies with the guidance of RG 1.136 (Reference 3.8-3). MHI Technical Report MUAP-10018 “US-APWR Containment Performance for Pressure Loads” (Reference 3.8-55) documents the methodology used to determine the pressure effects of an accident that

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		releases hydrogen generated from 100% fuel clad metal-water reaction and uncontrolled hydrogen burning on the PCCV. The maximum pressure considered in the analysis in MUAP-10018 (Reference 3.8-55) is P_g1 + P_g2 = 173.7 psia = 159 psig. The analysis also includes effects of dead load *D*." Reason: Update beyond DBA design approach [RAI 490-3732, Question 03.08.01-2]
3.8-11 to 12	Subsection 3.8.1.4.3	Replace Subsection in its entirety: "The US-APWR ultimate pressure capacity analyses are based on detailed 3D finite element modeling, advanced material constitutive relations including material degradation with temperature, and an assessment of uncertainties within a probabilistic framework. Accident conditions leading to over-pressurization include elevated temperatures. Because of thermal induced stresses and material property degradation at elevated temperatures, the fragility for over-pressurization is also a function of temperature. Thus, the fragility analyses are conducted for three different thermal conditions, 1) normal operating steady-state conditions, 2) a long term accident condition, and 3) a hydrogen burning condition. The analyses indicate that the pressure capacity is limited by liner tearing, which is found to first initiate at the transition to the thickened concrete section for the equipment hatch. The expected or median pressure to initiate tearing is found to be 223.6 psig or 3.29 times the design pressure (Pd) of 68 psig for the steady state thermal conditions associated with a long term accident condition. This limitation in pressure capacity due to liner tearing is consistent with the ¼ scale PCCV tests performed at Sandia National Laboratories (SNL), References 3.8-56 and 3.8-57. The 95% confidence value for liner tearing under long term accident conditions is determined to be 176 psig or 2.59* Pd in these analyses. The median capacity due to liner tearing for the hydrogen burning case is found to be 238.5 psig or 3.51* Pd. This pressure is higher than that at normal operating conditions, which is attributed to the compressive stress induced into the liner due to the locally higher temperatures of the liner relative to the concrete. However, note that the 95% high confidence value for pressure capacity due to liner tearing under hydrogen

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		burning conditions is lower than that for normal operating conditions reflecting the additional uncertainty for the severe accident conditions and effects of high temperatures. For ultimate capacity based on rebar and tendon rupture, the median pressure capacity for long term design accident conditions is found to be 243.6 psig or 3.58* Pd. It is also determined that the ultimate capacity is not limited by the concrete strength. These results are again consistent with the SNL test for the ¼ scale PCCV model. These analyses also indicate that the ultimate capacity does not strongly depend on temperature. The median ultimate capacity at normal operating temperature is determined to be 3.65*Pd and the median ultimate capacity under hydrogen burning conditions is 3.60*Pd. The fragility analyses and detailed description of the methodologies are summarized in MUAP-10018 (Reference 3.8-55).” Reason: Update beyond DBA design approach [RAI 490-3732, Question 03.08.01-2]
3.8-47	Subsection 3.8.4.1.3 1st Paragraph 4th Sentence	This portion is modified by designated as CDI: Change: “The UHSRS consist of a cooling tower enclosure, ESWS pump houses, and the UHS basin.” to “[[The UHSRS consist of a [[cooling tower enclosure]], ESWS pump houses, and the UHS [[basin]].]]” Reason: Design of components in UHSRS are site-specific, and design of UHS depends on site conditions.
3.8-57	Subsection 3.8.4.4.1 4th Paragraph 3rd Sentence	Change: “These loads are applied to the linear elastic FE model fixed at elevation 3 ft, 7 in. as equivalent static forces. Loads and load combinations are given in Subsection 3.8.4.3.” to “These loads are applied to the linear elastic FE model, which extends to the base of the R/B foundation, as equivalent static forces. Loads and load combinations are given in Subsection 3.8.4.3.” Reason: Design/analysis update
3.8-57	Subsection 3.8.4.4.1 4th Paragraph 4th Sentence	Add as 4th Sentence: “Soil stiffnesses derived from the standard plant soil profiles are assigned to the subgrade for the design of the overall R/B, and the design of the R/B superstructure is also performed considering a fixed-base condition at the bottom of the foundation” Reason: Provided clarification

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.8-70	Subsection 3.8.5.1.1 1st Paragraph 4th and 5th Sentences	Change: "The length of the basemat in the north-south direction is 309 ft, 0 in., and in the east-west direction is 210 ft, 0 in. The central region, with a diameter of approximately 188 ft, 0 in., supports the PCCV and containment internal structure with a thickness varying from 11 ft, 7 in. to 38 ft, 2 in. The peripheral portion which supports the R/B is 9 ft, 11 in. thick." to "The length of the basemat in the north-south direction is 309 ft, 0 in., and in the east-west direction is 210 ft, 0 in, as shown in Figure 3J-1. The central region, generally circular with a diameter of approximately 187 ft, supports the PCCV and containment internal structure with a thickness of approximately 38 ft, 2 in. The peripheral portion which supports the R/B is 9 ft, 11 in. thick." Reason: Update the description of the basemat [RAI 657-5135, Question 03.08.05-38]
3.8-71	Subsection 3.8.5.1.2 1st Paragraph 2nd Sentence	Change: "Each PS/B basemat is a rectangular reinforced concrete mat with a thickness of 100 in." to "Each PS/B basemat is a rectangular reinforced concrete mat with a thickness of 119 in." Reason: Editorial Correction
3.8-72	Subsection 3.8.5.4 1st Paragraph Last two Sentences	Delete last two sentences in their entirety. Reason: Update to be consistent with current design and analysis approaches.
3.8-74	Subsection 3.8.5.4.2.1 1st Paragraph 1st Sentence	Change: "The stress conditions of the basemat are generated by numerous types of loads from the superstructure." to "The stress conditions of the basemat for the R/B complex are generated by numerous types of loads from the superstructure." Reason: Provided clarification
3.8-74	Subsection 3.8.5.4.2.1 2nd to Last Paragraphs	Change in its entirety to: "Regarding the R/B, the element divisions in a horizontal direction inside the secondary shield walls of the containment internal structure are made in a rectangular grid pattern and those divisions outside the secondary shield wall are made in a polar pattern. Peripheral areas of the basemat, outside the thickened mat that supports the PCCV and containment internal structure are divided into a rectangular grid.

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		The upper portion of tendon gallery is considered with concentrated stresses created by the connection with the PCCV. This region is divided into multiple layers of elements in the radial direction to accommodate the differing concrete strengths in this area as shown schematically in Figure 3.8.5-4. The basemat below the PCCV and the lower portion of containment internal structure are simulated with solid elements (ANSYS SOLID45 elements). The elements below the PCCV are divided into ten layers and elements in peripheral areas are divided into four layers. The FE modeling of the PS/Bs is addressed in Subsection 3.8.4.4." Reason: Provided clarification
3.8-74	Subsection 3.8.5.4.3 1st Paragraph	Change: "The basemat subgrade in the FE model is represented by translational spring elements that are attached to the bottom of the basemat. The stiffness of the backfill around the below-grade walls is not considered in the model. Subgrade coefficients, determined based on the SSI lumped parameter values listed in Table 3H.2-14 of Appendix 3H, are used to assign spring values to the individual nodes of the FE model. These subgrade coefficients are multiplied by the basemat nodal point tributary areas to compute the spring constants assigned to the nodal points. The vertical spring stiffnesses are also developed in a manner such that the cumulative vertical stiffness is equivalent to the vertical SSI spring constant value in Table 3H.2-14." to "The basemat subgrade is included in the detailed static FE models used for structural design by meshing a sufficiently large volume of soil/rock below and around the basemat. The stiffness of the backfill around the below-grade walls is not considered in the model. To increase computational efficiency, the subgrade part of the FE model is condensed into a super-element. The properties of the subgrade layers used in the FE model of the subgrade are established based on several profiles selected from the generic layered soil profiles described in Technical Report MUAP-10001 (Reference 3.7-47) to cover the entire range of soil/rock conditions at representative nuclear power plant sites within the central and eastern US." Reason: Provided clarification that a representative volume of the subgrade will be included in the ANSYS

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		static models. [RAI 661-5129, Question 03.08.01-24]
3.8-77	Subsection 3.8.5.5.2 Last Paragraph	Add as last paragraph: "When a coefficient of friction of 0.7 is used in calculating sliding resistance F_s, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required." Reason: If a coefficient of friction of less than 0.7 is justified by the COL Applicant, "roughening" of the concrete is not required. [RAI 657-5135, Question 03.08.05-41]
3.8-80	Subsection 3.8.6 COL 3.8(30)	Add new COL item: "3.8(30) When a coefficient of friction of 0.7 is used in calculating sliding resistance F_s, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required." Reason: If a coefficient of friction of less than 0.7 is justified by the COL Applicant, "roughening" of the concrete is not required. [RAI 657-5135, Question 03.08.05-41]
3.8-81	Subsection 3.8.7 Reference 3.8-16	Change in its entirety to "Deleted." Reason: Provide consistency with design and analysis approaches
3.8-84	Subsection 3.8.7 References 3.8-55 to 3.8-57	Add 3 new references: "3.8-55 <u>US-APWR Containment Performance for Pressure Loads</u>, MUAP-10018, Mitsubishi Heavy Industries, Ltd, June, 2010. 3.8-56 <u>Pretest Analysis of a 1:4-Scale Prestressed Concrete Containment Vessel Model</u>, Dameron, R. A., Zhang, L., Rashid, Y. R., Vargas, M. S., NUREG/CR-6685, U. S. Nuclear Regulatory Commission, Washington, D. C., October 2000. 3.8-57 <u>Posttest Analysis of the NUPEC/NRC 1:4-Scale Prestressed Concrete Containment Vessel Model</u>, Dameron, R. A., Hansen, B. E., Parker, D. R., Rashid, Y. R., NUREG/CR-

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		6809, U. S. Nuclear Regulatory Commission, Washington, D. C., February 2003." Reason: Provide consistency with design and analysis approaches
3.8-102	Table 3.8.4-5 2nd Row	Delete 2nd Row Reason: Design/analysis update
3.8-102	Table 3.8.4-5 3rd Row 3rd Column	Change: To obtain member forces for thermal load" to "To obtain member forces including thermal load" Reason: Design/analysis update
3.8-187	Figure 3.8.3-6 Sheet 6	Replace figure in its entirety. Reason: Deleted machine room and changed sump shape.
3.9-14	Subsection 3.9.1.2.1 Last Bullet	Delete the bullet for SQUIRT Reason: PICEP is now used for LBB evaluation instead of SQUIRT.
3.9-18	Subsection 3.9.2.2.1 1st Paragraph 2nd Sentence	Change: "…ANSI/IEEE Std 344-1987 (Reference 3.9-15), as endorsed by NRC, RG 1.100, Rev. 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Reference 3.9-16)." to "…ANSI/IEEE Std 344-2004 (Reference 3.9-34), as endorsed by NRC, RG 1.100, Rev. 3, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Reference 3.9-16)." Reason: RG 1.100, Revision 3 Update
3.9-19	Subsection 3.9.2.2.2 1st Paragraph 3rd Sentence	Change: "The method of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Std 344-1987 (Reference 3.9-15) and Subsections 3.10.2 and 3.10.3." to "The method of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Std 344-2004 (Reference 3.9-34) and Subsections 3.10.2 and 3.10.3." Reason: RG 1.100, Revision 3 Update

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3.9-56	Subsection 3.9.4.1 1st Paragraph 1st Sentence	Change: "The control rod drive system (CRDS) provides one of the independent reactivity control systems, driving a rod control cluster assembly (RCCA)…" to "The control rod drive system (CRDS) provides one of the independent reactivity control systems, driving a rod cluster control assembly (RCCA)…" Reason: Editorial correction to modify the acronym of RCCA (rod cluster control assembly)
3.9-78	Subsection 3.9.6.1 4th Paragraph 1st Sentence	Change: "…the OM Code, 12 months before the date of issuance of the operating license and, in compliance with Plant, Technical Specification and this DCD." to "the OM Code, 12 months before initial fuel load and, in compliance with plant Technical Specifications and this DCD." Reason: Editorial correction to text that was revised to be consistent with 10 CFR 50.55a for Part 52 COL Applications.
3.9-90	Subsection 3.9.10 Reference 3.9-11	Change: "<u>Seepage Quantification of Upset in Reactor Tubes [SQUIRT], Code System to Predict Leakage Rate and Area of Crack Opening for Cracked Pipe in Nuclear Power Plants, Version 1.1</u>, Oak Ridge National Laboratory (PSR-533), Oak Ridge, TN, 2003." to "Deleted" Reason: PICEP is now used for LBB evaluation instead of SQUIRT.
3.9-91	Subsection 3.9.10 Reference 3.9-15	Change: "<u>IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations</u>, Institute of Electrical and Electronics Engineers, IEEE Std 344-1987." to "Deleted" Reason: RG 1.100, Revision 3 Update
3.9-91	Subsection 3.9.10 Reference 3.9-16	Change: "<u>Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants</u>. Regulatory Guide 1.100, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 1988." to "<u>Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants</u>. Regulatory Guide 1.100, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, September 2009." Reason: RG 1.100, Revision 3 Update

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3.9-92	Subsection 3.9.10 Reference 3.9-34	Change: "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std. 344-2004, Appendix D, Institute of Electrical and Electronics Engineers Power Engineering Society, New York, New York, June 2005." to "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std. 344-2004, Institute of Electrical and Electronics Engineers Power Engineering Society, New York, New York, June 2005." Reason: RG 1.100, Revision 3 Update
3.9-95	Subsection 3.9.10 Reference 3.9-65	Add new reference: "PICEP: Pipe Crack Evaluation Program. NP-3596-SR, Rev.1, Electric Power Research Institute, 1987." Reason: PICEP is now used for LBB evaluation instead of SQUIRT.
3.9-98	Table 3.9-3 Title	Change: "Minimum Design Loading Combinations for ASME Code, Section III, Class 1, 2, 3 and CS Systems and Components" to "Design Loading Combinations for ASME Code, Section III, Class 1, 2, 3 and CS Systems and Components" Reason: Editorial correction
3.9-98	Table 3.9-3 Note 11 2nd Sentence	Change: "If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading are 300 as derived in accordance with Institute of Electrical and Electronic Engineers Standard 344-1987 (Reference 3.9-15)." to "If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading are 300 as derived in accordance with Institute of Electrical and Electronic Engineers Standard 344-2004 (Reference 3.9-34)." Reason: RG 1.100, Revision 3 Update
3.9-99	Table 3.9-4 Title	Change: "Minimum Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components" to "Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components" Reason: Editorial Correction

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.9-99	Table 3.9-4 5^{th} Row	Change: “TH_i” to “TH_E” Reason: Clarify thermal load application for level C (Emergency) service.
3.9-99	Table 3.9-4 8^{th} Row	Change: “TH_i” to “TH_F” Reason: Clarify thermal load application for level D (Faulted) service.
3.9-100	Table 3.9-5 8^{th} and 9^{th} Rows	Add rows for “TH_E ASME Service Level C (Emergency) Thermal Load” and “TH_F ASME Service Level D (Faulted) Thermal Load” Reason: Clarify thermal load application for Emergency and Faulted Levels.
3.9-109	Table 3.9-11 Note 4 2^{nd} Sentence	Change: “The number of cycles to be considered for earthquake loading are 300 as derived in accordance with Appendix D of Institute of Electrical and Electronic Engineers Standard 344-1987 (Reference 3.9-16).” to “The number of cycles to be considered for earthquake loading are 300 as derived in accordance with Institute of Electrical and Electronic Engineers Standard 344-2004 (Reference 3.9-34).” Reason: RG 1.100, Revision 3 Update
3.9-126	Table 3.9-14 Sheet 7 3^{rd} Row	Delete row for “RCS-VLV-167” Reason: Correct erroneous addition
3.9-134	Table 3.9-14 Sheet 15 2^{nd} Row	Add “Transfer Close” to the 4^{th} Column Reason: Correct erroneous omitted information
3.9-136	Table 3.9-14 Sheet 17 2^{nd} Row	Change: “CVS-LCV121B” to “CVS-LCV-031B” Reason: Correct description of valve tag number
3.9-136	Table 3.9-14 Sheet 17 3^{rd} Row	Change: “CVS-LCV121C” to “CVS-LCV-031C” Reason: Correct description of valve tag number

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3.9-136	Table 3.9-14 Sheet 17 4th Row	Change: "CVS-LCV121D" to "CVS-LCV-031D" Reason: Correct description of valve tag number
3.9-136	Table 3.9-14 Sheet 17 5th Row	Change: "CVS-LCV121E" to "CVS-LCV-031E" Reason: Correct description of valve tag number
3.9-137	Table 3.9-14 Sheet 18 2nd Row	Change: "CVS-LCV121F" to "CVS-LCV-031F" Reason: Correct description of valve tag number
3.9-137	Table 3.9-14 Sheet 18 3rd Row	Change: "CVS-LCV121G" to "CVS-LCV-031G" Reason: Correct description of valve tag number
3.9-137	Table 3.9-14 Sheet 18 Last Row	Add in the 4th Column: "Transfer Close" Reason: Correct erroneously omitted information
3.9-138	Table 3.9-14 Sheet 19 3rd Row	Add new row for "CVS-VLV-594" Reason: Correct erroneously omitted information
3.9-141	Table 3.9-14 Sheet 22 3rd-5th Rows	Delete rows for "CVS-VLV-653", "CVS-VLV-667" and "CVS-VLV-667B" Reason: Correct erroneous addition
3.9-142	Table 3.9-14 Sheet 23 2nd and 3rd Rows	Delete rows for "CVS-VLV-667C" and "CVS-VLV-667D" Reason: Correct erroneous addition
3.9-143	Table 3.9-14 Sheet 24 7th and 8th Rows	Add in the 7th Column: "Exercise Full Stroke/Quarterly Operability Test" Reason: Correct erroneously omitted information
3.9-144	Table 3.9-14	Add in the 7th Column: "Exercise Full Stroke/Quarterly

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	Sheet 25 2nd and 3rd Rows	Operability Test" Reason: Correct erroneously omitted information
3.9-144	Table 3.9-14 Sheet 25 4th-6th Rows	Add in the 7th Column: "Check Exercise/Refueling Outage" Reason: Correct erroneously omitted information
3.9-145	Table 3.9-14 Sheet 26 2nd Row	Add in the 7th Column: "Check Exercise/Refueling Outage" Reason: Correct erroneously omitted information
3.9-145	Table 3.9-14 Sheet 26 3rd-5th Rows	Add in the 4th Column: "Transfer Open" Reason: Correct erroneously omitted information
3.9-146	Table 3.9-14 Sheet 27 2nd Row	Add in the 4th Column: "Transfer Open" Reason: Correct erroneously omitted information
3.9-157	Table 3.9-14 Sheet 38 4th-7th Rows	Delete the 4 rows for valves "SIS-VLV-058A", "SIS-VLV-058B", "SIS-VLV-058C" and "SIS-VLV-058D" Reason: Correct erroneous addition
3.9-158	Table 3.9-14 Sheet 39 2nd Row	Delete the row for valve "SIS-VLV-156" Reason: Correct erroneous addition
3.9-165	Table 3.9-14 Sheet 46 5th Row	Add in the 4th Column: "Transfer Close" Reason: Correct erroneously omitted information
3.9-166	Table 3.9-14 Sheet 47 2nd-5th Rows	Add in the 4th Column: "Transfer Close" Reason: Correct erroneously omitted information
3.9-167	Table 3.9-14 Sheet 48	Add in the 4th Column: "Transfer Close" Reason: Correct erroneously omitted information

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	2nd-4th Rows	
3.9-170	Table 3.9-14 Sheet 51 Last Row	Delete the row for valve "RHS-VLV-062A" Reason: Correct erroneous addition
3.9-171	Table 3.9-14 Sheet 52 2nd-4th Rows	Delete the 3 rows for valves "RHS-VLV-062B", "RHS-VLV-062C" and "RHS-VLV-062D" Reason: Correct erroneous addition
3.9-174	Table 3.9-14 Sheet 55 2nd Row	Delete in the 4th Column: "Maintain Close" Reason: Correct erroneously omitted information
3.9-174	Table 3.9-14 Sheet 55 4th and 5th Rows	Add in the 4th Column: "Maintain Open" Reason: Correct erroneously omitted information
3.9-175	Table 3.9-14 Sheet 56 2nd and 3rd Rows	Add in the 4th Column: "Maintain Open" Reason: Correct erroneously omitted information
3.9-175	Table 3.9-14 Sheet 56 4th and 5th Rows	Delete in the 4th Column: "Transfer Close" Reason: Correct erroneously omitted information
3.9-177	Table 3.9-14 Sheet 58 2nd-8th Rows	Add in the 4th Column: "Maintain Close" Reason: Correct erroneously omitted information
3.9-178	Table 3.9-14 Sheet 59 2nd Row	Add in the 4th Column: "Maintain Close" Reason: Correct erroneously omitted information
3.9-178	Table 3.9-14 Sheet 59 3rd and 4th Rows	Add new rows for "EFS-VLV-109B" and "EFS-VLV-109C" Reason: Engineering design/analysis development.

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3.9-179	Table 3.9-14 Sheet 60 2nd-5th Rows	Change in the 1st Column: “NFS” to “FWS” Add in the 4th Column: “Maintain Close” Reason: Correct erroneously omitted information
3.9-179	Table 3.9-14 Sheet 60 3rd Row	Change in the 1st Column: “NFS” to “FWS” Reason: Editorial correction
3.9-180	Table 3.9-14 Sheet 61 2nd and 3rd Rows	Change in the 1st Column: “NFS” to “FWS” Reason: Editorial correction
3.9-197	Table 3.9-14 Sheet 78 3rd-6th Rows	Change in the 6th Column: “BC” to “AC” Reason: Correct description of ASME IST Category
3.9-198	Table 3.9-14 Sheet 79 2nd-5th Rows	Delete the 4 rows for valves “CSS-VLV-023A”, “CSS-VLV-023B”, “CSS-VLV-023C” and “CSS-VLV-023D” Reason: Correct erroneous addition
3.9-221	Table 3.9-14 Sheet 102 5th and 6th Rows	Delete rows for valves “NCS-VLV-452A” and “NCS-VLV-452B” Reason: Correct erroneous addition
3.9-232	Table 3.9-14 Sheet 113 3rd and 4th Rows	Add as new 3rd Row: “PSS-MOV-052C”; “Containment spray/residual heat removal heat exchanger downstream sampling line isolation”; “Remote MO Globe”; “Maintain Close Transfer Close”; “Active Remote Position”; “B”; “Remote Position Indication, Exercise/2 Years Exercise Full Stroke/ Quarterly Operability Test” Add as new 4th Row: “PSS-MOV-052D”; “Containment spray/residual heat removal heat exchanger downstream sampling line isolation”; “Remote MO Globe”; “Maintain Close Transfer Close”; “Active Remote Position”; “B”; “Remote Position Indication, Exercise/2 Years Exercise Full Stroke/ Quarterly Operability Test” Reason: Correct erroneously omitted information.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.9-234	Table 3.9-14 Sheet 115 Last Row	Delete row for valve “PSS-VLV-091” Reason: Correct erroneous addition
3.9-241	Table 3.9-14 Sheet 122	Delete row for valve “RWS-VLV-073” Reason: Correct erroneous addition
3.9-242	Table 3.9-14 Sheet 123 2nd and 3rd Rows	Add Rows for “RWS-VLV-012A” and “RWS-VLV-012B” Reason: Correct erroneously omitted information
3.9-242	Table 3.9-14 Sheet 123 Last Row	Delete row for valve “CAS-VLV-004” Reason: Correct erroneous addition
3.9-243	Table 3.9-14 Sheet 124 Last Row	Delete row for valve “DWS-VLV-006” Reason: Correct erroneous addition
3.9-247	Table 3.9-14 Sheet 128 3rd and 4th Rows	Delete rows for valves “FSS-VLV-002” and “FSS-VLV-005” Reason: Correct erroneous addition
3.9-265	Table 3.9-14 Sheet 146 2nd Row	Delete row for valve “RMS-VLV-004” Reason: Correct erroneous addition
3.10-3	Subsection 3.10.1.1 3rd Paragraph	Change: “The seismic qualification and documentation procedures used for safety-related mechanical and electrical equipment and their supports are in accordance with the “IEEE Recommended Practice for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations”, ANSI/IEEE Std 344-1987 (Reference 3.10-6), as endorsed by the NRC, RG 1.100, Revision 2, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (Reference 3.10-7).” to “The seismic qualification and documentation procedures used for safety-related

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		mechanical and electrical equipment and their supports are in accordance with the "IEEE Recommended Practice for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations", ANSI/IEEE Std 344-2004 (Reference 3.10-8), as endorsed by the NRC, RG 1.100, Revision 3, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Reference 3.10-7)." Reason: RG 1.100 Revision 3 Update
3.10-3	Subsection 3.10.1.1 4th Paragraph	Change: "The US-APWR mechanical and electrical equipment seismic qualification meets IEEE Std 344-1987 (Reference 3.10-6) as modified by RG 1.100 (Reference 3.10-7) for qualification by either analysis, testing or by a combination of both testing and analysis, and as supplemented with the "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", IEEE Std 344-2004 (Reference 3.10-8) for use to seismically qualify equipment by an experience-based approach. IEEE Std 344-2004 (Reference 3.10-8) is to be endorsed by RG 1.100 (Reference 3.10-7) in a future revision as indicated in "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment", NUREG-0800, SRP 3.10 (Reference 3.10-9). Experience-based qualification is not used for any equipment." to "The US-APWR mechanical and electrical equipment seismic qualification meets IEEE Std 344-2004 (Reference 3.10-8) as modified by RG 1.100 (Reference 3.10-7) for qualification by either analysis, testing or by a combination of both testing and analysis." Reason: RG 1.100 Revision 3 Update
3.10-3	Subsection 3.10.1.1 5th Paragraph	Change: "The qualification of the design of safety-related, seismic category I mechanical equipment to assure the structural integrity of pressure boundary components follows the guidance provided in the ASME Boiler and Pressure Vessel Code, Section III (Reference 3.10-10). The US-APWR implements an operability program for active valves following the guidance in "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants", RG 1.148 (Reference 3.10-11) as discussed in Subsections 3.9.3 and 3.9.6." to "The qualification of the design of safety-related, seismic category I mechanical equipment to assure the

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		structural integrity of pressure boundary components follows the guidance provided in the ASME Boiler and Pressure Vessel Code, Section III (Reference 3.10-10). The US-APWR implements an operability program for active valves following the guidance in "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants", ASME QME-1-2007 (Reference 3.10-12) as discussed in Subsections 3.9.3 and 3.9.6." Reason: RG 1.100 Revision 3 Update
3.10-4	Subsection 3.10.2 1st Paragraph 1st Sentence	Change: "The recommended guidance and requirements in IEEE Std 344-1987 (Reference 3.10-6) and RG 1.100 (Reference 3.10-7) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment." to "The recommended guidance and requirements in IEEE Std 344-2004 (Reference 3.10-8) and RG 1.100 (Reference 3.10-7) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment." Reason: RG 1.100 Revision 3 Update
3.10-4	Subsection 3.10.2 1st Paragraph 2nd Sentence	Change: "The methods and guidance in "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants", ASME QME-1-2007 (Reference 3.10-12), including Appendix QR-A with exceptions to be provided in a future revision of RG 1.100 (Reference 3.10-7), are also used for seismic qualification of active mechanical equipment." to "The methods and guidance in "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants", ASME QME-1-2007 (Reference 3.10-12), including Appendix QR-A, with exceptions provided in RG 1.100 (Reference 3.10-7), are used for seismic qualification of active mechanical equipment." Reason: RG 1.100 Revision 3 Update
3.10-5	Subsection 3.10.2 5th Paragraph 3rd Sentence	Change: "High frequency failures resulting from improper design of mounting, inadequate design connections and fasteners, mechanical misalignment/binding of parts and the rare case of failure of a component part, will result from the same structural failure modes as those experienced during low frequency content spectra qualification testing in accordance with IEEE Std 344-1987 (Reference 3.10-6)." to "High frequency failures resulting from improper

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		design of mounting, inadequate design connections and fasteners, mechanical misalignment/binding of parts and the rare case of failure of a component part, will result from the same structural failure modes as those experienced during low frequency content spectra qualification testing in accordance with IEEE Std 344-2004 (Reference 3.10-8)." Reason: RG 1.100 Revision 3 Update
3.10-6	Subsection 3.10.2 14th Paragraph 1st Sentence	Change: "With the elimination of the OBE from design considerations, two alternatives exist that essentially maintain the requirements provided in IEEE Std 344-1987 (Reference 3.10-6) to qualify equipment…" to "With the elimination of the OBE from design considerations, two alternatives exist that essentially maintain the requirements provided in IEEE Std 344-2004 (Reference 3.10-8) to qualify equipment…" Reason: RG 1.100 Revision 3 Update
3.10-7	Subsection 3.10.2 16th Paragraph 1st Sentence	Change: "Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five 1/2 SSE events when followed by one full SSE may be used in accordance with Appendix D of IEEE Std 344-1987 (Reference 3.10-6) and Figure D.1 of IEEE Std 344-2004 (Reference 3.10-8)." to "Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five 1/2 SSE events when followed by one full SSE may be used in accordance with Figure D.1 of IEEE Std 344-2004 (Reference 3.10-8)." Reason: RG 1.100 Revision 3 Update
3.10-7	Subsection 3.10.2 17th Paragraph 1st Sentence	Change: "…and IEEE Std 344-1987 (Reference 3.10-6)." to "…and IEEE Std 344-2004 (Reference 3.10-8)." Reason: RG 1.100 Revision 3 Update
3.10-7	Subsection 3.10.2 **Testing** 1st Paragraph 1st Sentence	Change: "The seismic qualification testing inputs and methods for qualification of mechanical and electrical equipment are performed in accordance with the guidelines provided in IEEE Std 344-1987, Section 7 (Reference 3.10-6)." to "The seismic qualification testing inputs and methods for qualification of mechanical and electrical equipment are performed in accordance with the guidelines provided in IEEE Std 344-2004, Section 8 (Reference 3.10-8)."

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Reason: RG 1.100 Revision 3 Update
3.10-9	Subsection 3.10.2 **Analysis** 1st Paragraph 1st Sentence	Change: “The seismic analysis methods used are performed in accordance with the guidelines in IEEE Std 344-1987, Section 6 (Reference 3.10-6).” to “The seismic analysis methods used are performed in accordance with the guidelines in IEEE Std 344-2004, Section 7 (Reference 3.10-8).” Reason: RG 1.100 Revision 3 Update
3.10-9	Subsection 3.10.2 **Analysis** 1st Paragraph 2nd Sentence	Change: “…methods in accordance with IEEE Std 344-1987, Section 6 (Reference 3.10-6).” to “…methods in accordance with IEEE Std 344-2004, Section 7 (Reference 3.10-8).” Reason: RG 1.100 Revision 3 Update
3.10-10	Subsection 3.10.2 **Combined Testing and Analysis** 1st Paragraph 1st Sentence	Change: “The methods used for combined testing and analysis are performed in accordance with the guidelines in IEEE Std 344-1987, Section 8 (Reference 3.10-6).” to “The methods used for combined testing and analysis are performed in accordance with the guidelines in IEEE Std 344-2004, Section 9 (Reference 3.10-8).” Reason: RG 1.100 Revision 3 Update
3.10-11	Subsection 3.10.2.1.1 2nd Paragraph	Change: “…in accordance with the guidelines in IEEE Std 344-1987, Section 8 (Reference 3.10-6).” to “…in accordance with the guidelines in IEEE Std 344-2004, Section 9 (Reference 3.10-8).” Reason: RG 1.100 Revision 3 Update
3.10-11	Subsection 3.10.2.1.1 4th Paragraph 1st Sentence	Change: “Single-frequency testing can be used for line-mounted equipment and other equipment as recommended by IEEE Std 344-1987 (Reference 3.10-6) and RG 1.100 (Reference 3.10-7).” to “Single-frequency testing can be used for line-mounted equipment and other equipment as recommended by IEEE Std 344-2004 (Reference 3.10-8) and RG 1.100 (Reference 3.10-7).” Reason: RG 1.100 Revision 3 Update
3.10-11	Subsection 3.10.2.1.1 4th Paragraph	Change: “The method for qualification of line-mounted equipment is performed in accordance with the guidance in IEEE Std 344-1987, Section 7.6.7 (Reference 3.10-6) and IEEE Std 382-1996

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Last Sentence	(Reference 3.10-17), with justification and test results provided in the equipment qualification file." to "The method for qualification of line-mounted equipment is performed in accordance with the guidance in IEEE Std 344-2004, Section 8.6.7 (Reference 3.10-8) and IEEE Std 382-1996 (Reference 3.10-17), with justification and test results provided in the equipment qualification file" Reason: RG 1.100 Revision 3 Update
3.10-14	Subsection 3.10.2.2 <u>**Valves**</u> 8th Paragraph	Change: "The procedures acceptable to the NRC for implementing the regulations with respect to the detailed specification of information pertinent to defining the operating requirements for valve assemblies whose safety-related function is to open, close, or regulate fluid flow are discussed in RG 1.148 (Reference 3.10-11) with supplemental information for application of "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard", ANSI N278.1-1975 (Reference 3.10-19). The functional specifications for valves are addressed in Section 3.9." to "The procedures acceptable to the NRC for implementing the regulations with respect to the detailed specification of information pertinent to defining the operating requirements for valve assemblies whose safety-related function is to open, close, or regulate fluid flow are discussed in ASME QME-1-2007 (Reference 3.10-12). The functional specifications for valves are addressed in Section 3.9." Reason: RG 1.100 Revision 3 Update
3.10-16	Subsection 3.10.2.3 1st Paragraph 1st Sentence	Change: "…as well as during an SSE seismic event, in accordance with IEEE Std 344-1987 (Reference 3.10-6)." to "…as well as during an SSE seismic event, in accordance with IEEE Std 344-2004 (Reference 3.10-8)." Reason: RG 1.100 Revision 3 Update
3.10-16	Subsection 3.10.2.4 1st Paragraph Last Sentence	Change: "…described in IEEE Std 344-1987 (Reference 3.10-6) and ASME Code, Section III (Reference 3.10-10)." to "…described in IEEE Std 344-2004 (Reference 3.10-8) and ASME Code, Section III (Reference 3.10-10)." Reason: RG 1.100 Revision 3 Update
3.10-20	Subsection 3.10.6	Change: "<u>IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power</u>

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Reference 3.10-6	Generating Stations. American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987." to "Deleted" Reason: RG 1.100 Revision 3 Update
3.10-20	Subsection 3.10.6 Reference 3.10-7	Change: "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants. Regulatory Guide 1.100, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 1988." to "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants. Regulatory Guide 1.100, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, September 2009." Reason: RG 1.100 Revision 3 Update
3.10-20	Subsection 3.10.6 Reference 3.10-11	Change: "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants. Regulatory Guide 1.148, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981." to "Deleted" Reason: RG 1.100 Revision 3 Update
3.10-21	Subsection 3.10.6 Reference 3.10-19	Change: "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard. ANSI/ASME N278.1-1975 (Re-designated and Reaffirmed 1992), American National Standards Institute/American Society of Mechanical Engineers." to "Deleted" Reason: RG 1.100 Revision 3 Update
3.12-8	Subsection 3.12.4.1.1 Last Paragraph Last Bullet	Add new bullet: "• STAAD.Pro v8i STAAD.Pro is a general purpose structural analysis program. The program is used for steel stress analysis of piping supports." Reason: Provided information
3.12-18	Subsection 3.12.6.3.2 1st Paragraph 2nd Sentence	Change: "Support loads from these loading conditions are designated as TH_{MTL}, corresponding to the appropriate Level A, B, C, and D service conditions." to "Support loads from these loading conditions are designated as TH_{MTL}, TH_E and TH_F corresponding to the

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		appropriate Level A, B, C, and D service conditions." Reason: Clarify thermal load application for Emergency and Faulted Levels.
3.12-24	Subsection 3.12.8 Reference 3.12-18	Change: "PIPESTRESS, Piping Stress Analysis Program, Version 3.6.0." to "PIPESTRESS, Piping Stress Analysis Program, Version 3.6.2." Reason: Corrected version number.
3.12-25	Subsection 3.12.8 Reference 3.12-39	Change: "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, Appendix D, Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005." to "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005." Reason: RG 1.100 Revision 3 Update
3.12-26	Table 3.12-1 8th and 9th Rows	Add rows for "TH_E ASME Service Level C (Emergency) Thermal Load" and "TH_F ASME Service Level D (Faulted) Thermal Load" Reason: Clarify thermal load application for Emergency and Faulted Levels.
3.12-27	Table 3.12-2 Sheet 1 Note 3 2nd Sentence	Change: "If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading is to be 300 as derived in accordance with Appendix D of Institute of Electrical and Electronic Engineers Standard 344-2004 (Reference 3.12-39)." to "If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading is to be 300 as derived in accordance with the Institute of Electrical and Electronic Engineers Standard 344-2004 (Reference 3.12-39)." Reason: RG 1.100 Revision 3 Update
3.12-30	Table 3.12-4 4th Row	Change: "TH_{MTL}" to "TH_E" Reason: Clarify thermal load application for Level C Service.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.12-30	Table 3.12-4 5th Row	Change: "TH_{MTL}" to "TH_F" Reason: Clarify thermal load application for Level D Service.
3D-6 thru 3D-68	Table 3D-2 All Sheets 6th Column	Change Header of Purpose Column: "**RT, ESF, PAM, Pressure Boundary (PB), Other**" to "**RT, ESF, PAM, Pressure Boundary (PB), Other(1)**" Reason: Added Note 2 to provide clarification of requirements
3D-19	Table 3D-2 Sheet 15 Item 7	Change in the 2nd Column: "RMS-RE-091" to "RMS-RE-091A" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 8	Add Item 8 for component "RMS-RE-091B" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 9	Change in the 1st Column: "8" to "9" Change in the 2nd Column: "RMS-RE-092" to "RMS-RE-092A" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 10	Add Item 8 for component "RMS-RE-092B" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 11	Change in the 1st Column: "9" to "11" Change in the 2nd Column: "RMS-RE-093" to "RMS-RE-093A" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 12	Add Item 8 for component "RMS-RE-093B" Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2	Change in the 1st Column: "10" to "13"

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 15 Item 13	Change in the 2nd Column: “RMS-RE-094” to “RMS-RE-094A” Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 14	Add Item 8 for component “RMS-RE-094B” Reason: Editorial correction for inclusion of equipment
3D-19	Table 3D-2 Sheet 15 Item 15	Change in the 1st Column: “11” to “15” Reason: Editorial correction for addition of rows.
3D-33	Table 3D-2 Sheet 29 Items 41-42	Change in the 3rd Column: “Air Operated Valve” to “Level Control Valve” Reason: Correct valve type
3D-38	Table 3D-2 Sheet 34 Items 15-17	Change in the 8th Column: “Mild” to “Harsh” Reason: Correct description of Environmental condition
3D-39	Table 3D-2 Sheet 35 Item 18	Change: “Mild” to “Harsh” Reason: Correct description of Environmental condition
3D-39	Table 3D-2 Sheet 35 Item 23	Change in the 2nd Column: “EFS-MOV-103A” to “EFS-MOV-103A, EFS-MOV-103B” Change in the 3rd Column: “A-Emergency Feedwater Pump Actuation Valve” to “A-Emergency Feedwater Pump Actuation Valve on A-steam supply line, A-Emergency Feedwater Pump Actuation Valve on B-steam supply line” Reason: Corrected information
3D-39	Table 3D-2 Sheet 35 Item 24	Change in the 2nd Column: “EFS-MOV-103D” to “EFS-MOV-103C, EFS-MOV-103D” Change in the 3rd Column: “B-Emergency Feedwater Pump Actuation Valve” to “D-Emergency Feedwater Pump Actuation Valve on C-steam supply line, D-Emergency Feedwater Pump Actuation Valve on D-

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		steam supply line" Reason: Corrected information
3D-39	Table 3D-2 Sheet 35 Items 1-4	Change: "VLV" to "SMV" Reason: Correct valve type
3D-42	Table 3D-2 Sheet 38 Items 45-55 6th Column	Change: "ESF" to "Other" Reason: Correct description of Purpose
3D-43	Table 3D-2 Sheet 39 Items 56-59 6th Column	Change: "ESF" to "Other" Reason: Correct description of Purpose
3D-45	Table 3D-2 Sheet 41 Item 39	Change in the 5th Column: "7" to "13-3" Reason: Correct description of Location
3D-47	Table 3D-2 Sheet 43 Items 86-97	Add 12 Rows for NCS-MOV-321A, NCS-MOV-321B, NCS-MOV-322A, NCS-MOV-322B, NCS-MOV-323A, NCS-MOV-323B, NCS-MOV-324A, NCS-MOV-324B, NCS-MOV-325A, NCS-MOV-325B, NCS-MOV-326A, and NCS-MOV-326B. Reason: Correct erroneously omitted information
3D-48	Table 3D-2 Sheet 44 **Equipment (Spent Fuel Pit Cooling and Purification System)** Items 1-4 7th Column	Change: "2wks" to "1yr" Reason: Correct description of Operational Duration.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3D-49	Table 3D-2 Sheet 45 Items 9-10	Add 2 new rows for PSS-MOV-052C and PSS-MOV-052D Reason: Correct erroneously omitted information
3D-50	Table 3D-2 Sheet 46 Items 9-16 1st Column	Change Item numbers. “9” to “11”, “10” to “12”… “16” to “18” Reason: Editorial correction to fix numbering after insertion of two rows.
3D-50	Table 3D-2 Sheet 46 Items 15-16 12th Column	Change: “Non” to “I” Reason: Correct description of Seismic Category.
3D-52	Table 3D-2 Sheet 48 Items 23-26 and 29 3rd Column	Change: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3D-53	Table 3D-2 Sheet 49 Items 30-40 3rd Column	Change: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3D-54	Table 3D-2 Sheet 50 Items 5-10 3rd Column	Change: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3D-55 to 56	Table 3D-2 Sheets 51-52 Items 31-44	Add rows for “VRS-MEH-202A”, “VRS-MEH-202B”, “VRS-MEH-202C”, “VRS-MEH-202D”, “VRS-MEH-203A”, “VRS-MEH-203B”, “VRS-MEH-203C”, “VRS-MEH-203D”, “VRS-MEH-211A”, “VRS-MEH-211B”, “VRS-MEH-204A”, “VRS-MEH-204B”, “VRS-MEH-204C”, and “VRS-MEH-204D” Reason: Engineering design/analysis development.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3D-55 to 64	Table 3D-2 Sheets 51-60 Items 45-194	Change numbering to include the addition of rows. Reason: Editorial correction
3D-56	Table 3D-2 Sheet 52 Items 45-57 3rd Column	Change: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3D-56	Table 3D-2 Sheet 53 Items 58-60, and 69-76 3rd Column	Change: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3D-67	Table 3D-2 Sheet 63 Item 55	Change in the 8th Column: “Mild” to “Harsh” Reason: Correct Environmental Condition
3D-68	Table 3D-2 Sheet 64 Notes	Add note 2: “2. Identification number for “Purpose” (1) All active valves in Table 3D-2 have the function and operating duration of “PB-1yr” in addition to any other requirements..” Reason: Added Note 2 to provide clarification of requirements
3H-i	Contents	Change 3H.2 and 3H.3 to “Deleted” Delete Tables and Figures Reason: Engineering design/analysis development.
3H-v	Acronyms and Abbreviations	Delete “ASCE”, “FE”, “RCL”, “SSI”, and “SRSS”. Reason: Editorial, acronyms no longer used in appendix.
3H-1	Section 3H.1	Replace Section 3H.1 in its entirety with: “Refer to MUAP-10001, “Seismic Design Bases of the US-APWR

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Standard Plant” (Reference 3H-4) for the model properties for lumped mass stick model of reactor building (R/B)- prestressed concrete containment vessel (PCCV) – Containment Internal structure on a common basemat.” Reason: Engineering design/analysis development.
3H-1 to 3H-2	Section 3H.2 and Section 3H.3	Change Title to “Deleted” and delete all text in the sections. Reason: Engineering design/analysis development.
3H-3	Section 3H.4 References 3H-1, 3H-2 and 3H-3	Change to “Deleted” Reason: Engineering design/analysis development.
3H-3	Section 3H.4 Reference 3H-4	Add Reference 3H-4: “<u>Seismic Design Bases of the US-APWR Standard Plant</u>, MUAP-10001, Revision 2, Mitsubishi Heavy Industries, January 2011.” Reason: Engineering design/analysis development.
3H-4 to 39	All tables and figures	Delete Figures in their entirety. Reason: Engineering design/analysis development.
3I-1	Subsection 3I.1 1st Paragraph 1st Sentence	Change: “Refer to MUAP-08005, “Dynamic Analysis of the Coupled RCL-R/B-PCCV-CIS Lumped Mass Stick Model” (Reference 3I-1), and MUAP-08002, “Enhanced Information for PS/B Design” (Reference 3I-2), for in-structure response spectra (ISRS) for various buildings and elevations of the US-APWR.” to “Refer to MUAP-10006, “Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant” (Reference 3I-3) for the in-structure response spectra (ISRS) for various buildings and elevations of the US-APWR standard plant.” Reason: Updated References to be consistent with updates/additions of technical reports
3I-1	3I.2 References	Change 3I-1: “<u>Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model</u>, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008.” to “Deleted.” Change 3I-2: “Enhanced Information for PS/B Design,

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		MUAP-08002, Mitsubishi Heavy Industries, Ltd., February 2008." to "Deleted." Add 3I-3: "Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant, MUAP-10006, Rev.1, January 2011." Reason: Updated References to be consistent with updates/additions of technical reports
3J-1 to 2	Figure 3J-3 Figure 3J-4	Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected. Base mat thickness is corrected from 100" to 119.
3J-4	Figure 3J-1 Sheet 3	Modified figure to include the door opening. Reason: Detailed engineering progress
3J-6	Figure 3J-1 Sheet 5	Modified figure to change door opening location due to the change of panel arrangement. Reason: Detailed engineering progress
3J-10	Figure 3J-1 Sheet 9	Modified figure to change opening height from 7'-11" to 8'-7". Reason: Detailed engineering progress
3J-14	Figure 3J-1 Sheet 13	Modified figure to change opening height from 7'-11" to 8'-7" and added door opening. Reason: Detailed engineering progress
3K-25	Table 3K-2 Sheet 1 Items 4, 5, and 7-11	Change in the 7th Column: "FA2-117-24" to "FA2-127-08" Reason: Revised Fire Zone number
3K-25	Table 3K-2 Sheet 1 Item 16	Change in the 7th Column: "FA2-120-02" to "FA2-154-01" Reason: Revised Fire Zone number
3K-26	Table 3K-2	Change in the 7th Column: "FA2-120-06" to "FA2-154-

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 2 Item 18	05” Reason: Revised Fire Zone number
3K-26	Table 3K-2 Sheet 2 Item 29	Change in the 7th Column: “FA2-120-04” to “FA2-154-03” Reason: Revised Fire Zone number
3K-27	Table 3K-2 Sheet 3 Item 33	Change in the 7th Column: “FA2-120-06” to “FA2-154-15” Reason: Revised Fire Zone number
3K-27	Table 3K-2 Sheet 3 Items 35	Change in the 7th Column: “FA2-151-01” to “FA2-151-03” Reason: Revised Fire Zone number
3K-27	Table 3K-2 Sheet 3 Items 39	Change in the 7th Column: “FA2-152-01” to “FA2-152-03” Reason: Revised Fire Zone number
3K-27	Table 3K-2 Sheet 3 Item 41	Change in the 7th Column: “FA2-120-06” to “FA2-154-05” Reason: Revised Fire Zone number
3K-27	Table 3K-2 Sheet 3 Item 43	Change in the 7th Column: “FA2-120-02” to “FA2-154-01” Reason: Revised Fire Zone number
3K-28	Table 3K-2 Sheet 4 Item 49	Change in the 7th Column: “FA2-151-01” to “FA2-151-04” Change in the 8th Column: “N/A” to “above flood elevation” Change in the 9th Column: “-” to “0.69” Reason: Revised Fire Zone number
3K-28	Table 3K-2 Sheet 4 Item 50	Change in the 7th Column: “FA2-117-08” to “FA2-209-03” Reason: Revised Fire Zone number

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-28	Table 3K-2 Sheet 4 Item 54	Change in the 7th Column: "FA2-117-07" to "FA2-128-02" Reason: Revised Fire Zone number
3K-29	Table 3K-2 Sheet 5 Items 75 and 76	Change in the 7th Column: "FA2-120-06" to "FA2-154-05" Reason: Revised Fire Zone number
3K-29	Table 3K-2 Sheet 5 Items 79 and 80	Change in the 7th Column: "FA2-117-23" to "FA2-322-01" Reason: Revised Fire Zone number
3K-29	Table 3K-2 Sheet 5 Item 80	Add two rows after item 80 for "PSS-MOV-052C" and "PSS-MOV-052D" Reason: Engineering design/analysis development.
3K-30 to 44	Table 3K-2 Sheets 6-20 All items	Revise the Item number for the addition of two rows. Reason: Editorial correction
3K-30	Table 3K-2 Sheet 6 Items 89 and 90	Change in the 7th Column: "FA2-117-09" to "FA2-211-01" Reason: Revised Fire Zone number
3K-30	Table 3K-2 Sheet 6 Items 94 and 96	Change in the 7th Column: "FA2-117-32" to "FA2-416-01" Reason: Revised Fire Zone number
3K-30	Table 3K-2 Sheet 6 Items 95 and 97	Change in the 7th Column: "FA2-117-29" to "FA2-417-01" Reason: Revised Fire Zone number
3K-30	Table 3K-2 Sheet 6 Items 98 and 100	Change in the 3rd Column: "Motor Operated Damper" to "Electro Hydraulic Operated Damper" Change in the 7th Column: "FA2-117-32" to "FA2-416-01" Reason: Engineering design/analysis development

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		and Revised Fire Zone number
3K-30	Table 3K-2 Sheet 6 Item 99	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Change in the 7th Column: “FA2-117-29” to “FA2-417-01” Reason: Engineering design/analysis development and Revised Fire Zone number
3K-31	Table 3K-2 Sheet 7 Items 101 and 103	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Change in the 7th Column: “FA2-117-29” to “FA2-417-01” Reason: Engineering design/analysis development and Revised Fire Zone number
3K-31	Table 3K-2 Sheet 7 Item 102	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Change in the 7th Column: “FA2-117-32” to “FA2-416-01” Reason: Engineering design/analysis development and Revised Fire Zone number
3K-31	Table 3K-2 Sheet 7 Items 104, 108, and 112	Change in the 7th Column: “FA2-120-05” to “FA2-154-04” Reason: Revised Fire Zone number
3K-32	Table 3K-2 Sheet 8 Items 116, 120 and 124	Change in the 7th Column: “FA2-120-05” to “FA2-154-04” Reason: Revised Fire Zone number
3K-33	Table 3K-2 Sheet 9 Items 128, 130, 132, and 133	Change in the 7th Column: “FA2-117-32” to “FA2-416-01” Reason: Revised Fire Zone number
3K-33	Table 3K-2 Sheet 9	Change in the 7th Column: “FA2-117-29” to “FA2-417-01”

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Items 129, 131, 134, and 135	Reason: Revised Fire Zone number
3K-34	Table 3K-2 Sheet 10 Item 136	Change in the 7th Column: “FA2-117-32” to “FA2-416-01” Reason: Revised Fire Zone number
3K-34	Table 3K-2 Sheet 10 Item 137	Change in the 7th Column: “FA2-117-29” to “FA2-417-01” Reason: Revised Fire Zone number
3K-35	Table 3K-2 Sheet 11 Items 154 and 156	Change in the 7th Column: “FA2-117-34” to “FA2-409-02” Reason: Revised Fire Zone number
3K-35	Table 3K-2 Sheet 11 Items 155 and 157	Change in the 7th Column: “FA2-117-40” to “FA2-511-01” Reason: Revised Fire Zone number
3K-35	Table 3K-2 Sheet 11 Item 158	Change in the 7th Column: “FA2-117-90” to “FA2-209-04” Reason: Revised Fire Zone number
3K-35	Table 3K-2 Sheet 11 Item 159	Change in the 7th Column: “FA2-117-43” to “FA2-418-01” Reason: Revised Fire Zone number
3K-35	Table 3K-2 Sheet 11 Items 160 and 162	Change in the 7th Column: “FA2-120-06” to “FA2-154-05” Reason: Revised Fire Zone number
3K-36	Table 3K-2 Sheet 12 Item 164	Change in the 7th Column: “FA2-117-90” to “FA2-209-04” Reason: Revised Fire Zone number
3K-36	Table 3K-2 Sheet 12 Item 166	Change in the 7th Column: “FA2-117-90” to “FA2-209-03” Reason: Revised Fire Zone number

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-36	Table 3K-2 Sheet 12 Item 165	Change in the 7th Column: “FA2-117-43” to “FA2-418-01” Reason: Revised Fire Zone number
3K-36	Table 3K-2 Sheet 12 Item 169	Change in the 7th Column: “FA2-117-07” to “FA2-128-02” Reason: Revised Fire Zone number
3K-36	Table 3K-2 Sheet 12 Items 170 and 174	Change in the 7th Column: “FA2-120-04” to “FA2-154-03” Reason: Revised Fire Zone number
3K-36	Table 3K-2 Sheet 12 Item 176	Change in the 7th Column: “FA2-152-03” to “FA2-209-03” Reason: Revised Fire Zone number
3K-37	Table 3K-2 Sheet 13 Item 181	Change in the 7th Column: “FA2-117-07” to “FA2-128-02” Reason: Revised Fire Zone number
3K-37	Table 3K-2 Sheet 13 Items 182 and 183	Change in the 7th Column: “FA2-117-44” to “FA2-210-21” Reason: Revised Fire Zone number
3K-37	Table 3K-2 Sheet 13 Item 184	Change in the 7th Column: “FA2-117-05” to “FA2-154-04” Reason: Revised Fire Zone number
3K-37	Table 3K-2 Sheet 13 Item 188 and 189	Change in the 7th Column: “FA2-117-32” to “FA2-416-01” Reason: Revised Fire Zone number
3K-37	Table 3K-2 Sheet 13 Item 190 and 191	Change in the 7th Column: “FA2-117-29” to “FA2-417-01” Reason: Revised Fire Zone number
3K-37	Table 3K-2	Change in the 7th Column: “FA2-117-40” to “FA2-511-

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 13 and 14 Item 196 and 197	01” Reason: Revised Fire Zone number
3K-38	Table 3K-2 Sheet 14 Item 202 and 203	Change in the 7th Column: “FA2-117-42” to “FA2-209-05” Reason: Revised Fire Zone number
3K-39	Table 3K-2 Sheet 15 Item 217	Change in the 7th Column: “FA2-117-08” to “FA2-209-03” Reason: Revised Fire Zone number
3K-40	Table 3K-2 Sheet 16 Item 223	Change in the 7th Column: “FA2-117-07” to “FA2-128-02” Reason: Revised Fire Zone number
3K-41	Table 3K-2 Sheet 17 Item 232	Change in the 6th Column: “25’-3” ” to “76’-5” ” Change in the 7th Column: “FA2-151-05” to “FA2-506-01” Change in the 8th Column: “0.53” to “0.84” Reason: Revised Floor Elevation, Fire Zone number and Flood elevation above Floor.
3K-41	Table 3K-2 Sheet 17 Item 234	Change in the 6th Column: “76’-5” ” to “25’-3” ” Change in the 7th Column: “FA2-117-35” to “FA2-152-05” Change in the 8th Column: “0.88” to “0.69” Reason: Revised Floor Elevation, Fire Zone number and Flood elevation above Floor.
3K-41	Table 3K-2 Sheet 17 Item 235	Change in the 7th Column: “FA2-117-35” to “FA2-410-02” Change in the 8th Column: “0.88” to “0.99” Reason: Revised Fire Zone number and Flood elevation above Floor.
3K-41	Table 3K-2 Sheet 17 Item 236	Change in the 7th Column: “FA2-120-02” to “FA2-154-03” Reason: Revised Fire Zone number

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-41	Table 3K-2 Sheet 17 Item 237	Change in the 7th Column: “FA2-151-01” to “FA2-151-03” Reason: Revised Fire Zone number
3K-41	Table 3K-2 Sheet 17 Item 238	Change in the 2nd Column: “RHS-TE-024” to “RHS-TE-034” Change in the 7th Column: “FA2-152-01” to “FA2-152-03” Reason: Revised Equipment Tag and Fire Zone number
3K-41	Table 3K-2 Sheet 17 Item 239	Change in the 2nd Column: “RHS-TE-034” to “RHS-TE-044” Change in the 7th Column: “FA2-153-01” to “FA2-153-03” Reason: Revised Equipment Tag and Fire Zone number
3K-42	Table 3K-2 Sheet 17 and 18 Items 240, 241, 242	Change in the 7th Column: “FA2-120-06” to “FA2-154-05” Reason: Revised Fire Zone number
3K-42	Table 3K-2 Sheet 18 Items 252,253,254	Change in the 7th Column: “FA2-120-04” to “FA2-154-03” Reason: Revised Fire Zone number
3K-43	Table 3K-2 Sheet 19 Item 258	Change in the 2nd Column: “VRS-TS-526” to “VRS-TS-326” Reason: Editorial correction
3K-43	Table 3K-2 Sheet 19 Items 264,265,266	Change in the 7th Column: “FA2-117-32” to “FA2-416-01” Reason: Revised Fire Zone number
3K-43	Table 3K-2 Sheets 19 and 20 Items 267,268, 269	Change in the 7th Column: “FA2-117-29” to “FA2-417-01” Reason: Revised Fire Zone number

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-45	Table 3K-3 Sheet 1 Item 6	Change: "FA2-501-08" to "FA2-512-01" Reason: Revised Fire Zone number
3K-46	Table 3K-3 Sheet 2 Item 23	Change in the 2nd Column: "EFS-MOV-103A" to "EFS-MOV-103A, EFS-MOV-103B" Change in the 3rd Column: "A-Emergency Feedwater Pump Actuation Valve" to "A-Emergency Feedwater Pump Actuation Valve on A-steam supply line, A-Emergency Feedwater Pump Actuation Valve on B-steam supply line" Reason: Editorial Correction
3K-46	Table 3K-3 Sheet 2 Item 24	Change in the 2nd Column: "EFS-MOV-103D" to "EFS-MOV-103C, EFS-MOV-103D", Change in the 3rd Column: "B-Emergency Feedwater Pump Actuation Valve" to "D-Emergency Feedwater Pump Actuation Valve on C-steam supply line, D-Emergency Feedwater Pump Actuation Valve on D-steam supply line". Reason: Editorial Correction
3K-46 to 47	Table 3K-3 Sheets 2 and 3 Items 25-28	Change in the 2nd Column: "VLV" to "SMV" Reason: Editorial Correction
3K-47 to 48	Table 3K-3 Sheets 3 and 4 Items 29-52	Change in the 2nd Column: "VLV" to "SRV" Reason: Editorial correction
3K-49	Table 3K-3 Sheet 5 Items 61-64	Change in the 2nd Column: "AOV" to "SMV" Reason: Editorial Correction
3K-50	Table 3K-3 Sheet 6 Items 87-88	Change in the 2nd Column: "VLV" to "SRV" Reason: Editorial Correction
3K-51	Table 3K-3	Change in the 5th Column: "E" to "W"

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 7 Item 94	Reason: Editorial Correction
3K-51	Table 3K-3 Sheet 7 Item 96	Change in the 2nd Column: “NCS-LCV-1200” to “NCS-LCV-010” Change in the 7th Column: “FA2-601-01” to “FA2-603-01” Reason: Editorial Correction
3K-51	Table 3K-3 Sheet 7 Items 102	Change in the 7th Column: “FA2-602-01” to “FA2-604-01” Reason: Editorial Correction
3K-52	Table 3K-3 Sheet 8 Items 105-108	Change in the 2nd Column: “SSR” to “SST” Reason: Editorial Correction
3K-54	Table 3K-3 Sheet 10 Items 139 and 141	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Change in the 7th Column: “FA2-407-04” to “FA2-420-01” Reason: Engineering design/analysis development and Revised Fire Zone number
3K-54	Table 3K-3 Sheet 10 Items 140 and 142	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Change in the 7th Column: “FA2-407-01” to “FA2-423-01” Reason: Engineering design/analysis development and Revised Fire Zone number
3K-54	Table 3K-3 Sheet 10 Items 145-147	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.
3K-55	Table 3K-3 Sheet 11 Items 148-156	Change in the 3rd Column: “Motor Operated Damper” to “Electro Hydraulic Operated Damper” Reason: Engineering design/analysis development.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-57 to 58	Table 3K-3 Sheets 13-14 Items 187-200	Add rows for "VRS-MEH-202A", "VRS-MEH-202B", "VRS-MEH-202C", "VRS-MEH-202D", "VRS-MEH-203A", "VRS-MEH-203B", "VRS-MEH-203C", "VRS-MEH-203D", "VRS-MEH-211A", "VRS-MEH-211B", "VRS-MEH-204A", "VRS-MEH-204B", "VRS-MEH-204C", and "VRS-MEH-204D" Reason: Engineering design/analysis development.
3K-59 to 75	Table 3K-3 Sheets 15-31 Items 201-434	Change numbering to include the addition of rows. Reason: Editorial correction
3K-59	Table 3K-3 Sheet 15 Items 201-216	Change in the 3rd Column: "Motor Operated Damper" to "Electro Hydraulic Operated Damper" Reason: Engineering design/analysis development.
3K-75 , 78, and 79	Table 3K-4 Sheet 1, 4, 5 Items 1,5,9,49,71	Change in the 7th Column: "FA3-104-04" to "FA3-104-03" Reason: Revised Fire Zone number
3K-75, 78 and 80	Table 3K-4 Sheet 1,4,6 Items 4,8,12,52,74	Change in the 7th Column: "FA3-111-04" to "FA3-111-03" Reason: Revised Fire Zone number
3K-75	Table 3K-4 Sheet 1 Items 1-4	Change in the 2nd Column: "RFN" to "MFN" Reason: Editorial Correction
3K-75	Table 3K-4 Sheet 1 Items 5-12	Change in the 3rd Column: "Motor Operated Damper" to "Electro Hydraulic Operated Damper" Reason: Engineering design/analysis development.
3K-78	Table 3K-4 Sheet 4 Items 45-48	Change in the 2nd Column: "VLV" to "SRV" Reason: Editorial Correction
3K-82	Figure 3K-1	Change Layout to update wall configuration Reason: Detailed engineering progress

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3K-84	Figure 3K-3	Change Layout to add doors Reason: Detailed engineering progress
3K-85	Figure 3K-4	Change Layout to update wall configuration Reason: Detailed engineering progress
3K-86	Figure 3K-5	Change Layout to update wall configuration Reason: Detailed engineering progress
3K-87	Figure 3K-6	Change Layout to update wall configuration Reason: Detailed engineering progress
3K-88	Figure 3K-7	Change Layout to update wall configuration Reason: Detailed engineering progress
3K-89	Figure 3K-8	Change Layout to update elevation Reason: Detailed engineering progress
3K-93	Figure 3K-12	Change Layout to add doors Reason: Detailed engineering progress

APPENDICES

TABLES

Figures

ACRONYMS AND ABBREVIATIONS (Continued)

EFW	emergency feedwater
EFWS	emergency feedwater system
~~ELS~~	~~emergency letdown system~~
EPRI	Electrical Power Research Institute
EPS	emergency power source
EQ	environmental qualification
EQSDS	equipment qualification summary data sheet
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESQR	equipment seismic qualification report
ESWPT	essential service water pipe tunnel
ESWS	essential service water system
FE	finite element
FIRS	foundation input response spectra
FSS	Fire Protection Water Supply System
FW	feedwater
FWS	feedwater system
GA	general arrangement
GDC	General Design Criteria
GMRS	ground motion response spectra
GTG	gas turbine generator
HELB	high-energy line break
HHIS	high-head injection system
HIS	hydrogen ignition system
HMS	hydrogen monitoring system
HRC	rockwell c hardness
HSLA	high strength low alloy
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronic Engineers
ILRT	integrated leak rate test
ISI	inservice inspection
ISM	independent support motion
ISRS	in-structure response spectra
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
ITP	initial test program
LB	lower bound

ACRONYMS AND ABBREVIATIONS (Continued)

LOCA	loss-of-coolant accident
LOF	left-out-force
MCR	main control room
MELB	moderate-energy line break
MFIV	main feedwater isolation valve
MOV	motor operated valve
MS	main steam
MSS	main steam supply system
MT	magnetic particle examination method
MTC	moderator temperature coefficient
NCIG	National Construction Issues Group
NDE	nondestructive examination
NDRC	National Defense Research Council
NIST	National Institute of Standards and Technology
NPS	nominal pipe size
NRCA	non-radiological controlled area
NRC	U.S. Nuclear Regulatory Commission
OBE	operating-basis earthquake
OD	outside diameter
P&ID	piping and instrumentation diagram
PC	plant condition
PCCV	prestressed concrete containment vessel
PGA	peak ground acceleration
PIV	pressure isolation valve
PMF	probable maximum flood
PMP	probable maximum precipitation
PORV	power operated relief valve
POV	power operated valve
PS/B	power source building
PSFSV	power source fuel storage vault
PSMS	protection and safety monitoring system
PT	liquid penetrant examination method
PTFE	polytetra fluoroethylene
PWR	pressurized water reactor
QA	quality assurance
QAP	quality assurance program
R/B	reactor building
RCA	radiological controlled area
RCCA	rod cluster control ~~cluster~~ assembly

> leakage from the reactor coolant pressure boundary and rupture of small piping or other small components, which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

3.1.4.4.1 Discussion

The CVCS provides the normal means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the Volume Control Tank falls below a preset level. Centrifugal charging pumps (CCPs) are used as the normal means of reactor coolant makeup. The pumps are powered from the non-safety bus.

The CCPs are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from the non-safety bus. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure.

The emergency letdown system ~~(ELS)~~ consists of two emergency letdown lines from the RCS hot legs to the refueling water storage pit (RWSP). In the event that the normal CVCS letdown and boration capability is not available, the feed and bleed emergency letdown and boration operation can be utilized to achieve a cold shutdown boration level in the reactor coolant. The emergency letdown directs reactor coolant to the RWSP. The safety injection pumps (SIPs) provide borated coolant to the RCS from the RWSP. The SIPs are powered from a safety 1E bus so they can be supplied power from either the Offsite or Onsite Electric Power Systems.

Details of the system design, including the descriptions of the effects of small piping and component ruptures, are provided in Chapters 6, 9, and 15, and details of the electric power system are included in Chapter 8.

3.1.4.5 Criterion 34 – Residual Heat Removal

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. II

3.1.4.6.1 Discussion

The ECCS of the US-APWR includes the accumulator system, HHIS and emergency letdown system ~~ELS~~. The ECCS has the capability to mitigate the effects of any LOCA within the design bases. Cooling water is provided in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that clad metal-water reaction is limited to less than 1%. Design provisions assure performance of the required safety functions even with a postulated single failure.

Emergency core cooling is provided even if there is a failure of any component in the system. A passive system of four accumulators, one for each RCS loop, do not require any external signals or sources of power to operate, and provide the short-term cooling requirements for breaks in the large reactor coolant pipe systems. An independent and redundant pumping system is provided by the HHIS. The HHIS consists of four independent trains, each train contains a SIP and the associated valves, and piping. One of four independent safety electrical buses is available to each SIP. The SIPs are aligned to take suction from the RWSP that is inside containment to deliver borated water to the safety injection (SI) nozzles on the RV for short-term cooling and to the hot legs and downcomer for long-term cooling. Two SI trains are capable of meeting the design cooling function for a large LOCA assuming single failure in one train with another train out of service for maintenance.

The Discussion section of GDC 33 describes the emergency letdown system ~~ELS~~, the system's flow path, and boration capability via the SIPs, which are supplied power from the safety bus from either the offsite or onsite sources, as needed.

These systems are arranged so that a single failure of any active component does not interfere with meeting the short- term cooling requirements.

Additionally, the ECCS is designed with sufficient redundancy (four trains) to accomplish its safety functions assuming a single failure of an active component or a passive component in the long term following an accident with one train out of service for maintenance.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a LOCA. This limits the fuel-clad temperature; assures that the core will remain intact and in place, with its essential heat transfer geometry preserved; and prevents a return to criticality. This protection is provided for the following events:

- All pipe breaks sizes up to and including the hypothetical circumferential rupture of the largest pipe of a reactor coolant loop
- A LOCA associated with a rod ejection

The ECCS is described in Chapter 6. The LOCA, including an evaluation of consequences, is discussed in Chapter 15. Details of the electric power system are described in Chapter 8.

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 3 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Pressurizer piping downstream of the loop seal drain stop valve: RCS-VLV-157,158,159,160	4	PCCV	D	N/A	4	NS	
Pressurizer vent piping up to RCS-VLV-153 and the valve RCS-VLV-153	2	PCCV	B	YES	2	I	
Auxiliary spray TC piping up to RCS-VLV-151 and the valve RCS-VLV-151	2	PCCV	B	YES	2	I	
Reactor vessel head vent piping upstream of and including the reactor vessel head vent valves RCS-MOV-002A, B,003A,B	1	PCCV	A	YES	1	I	
Reactor vessel head vent line piping downstream of and excluding the reactor vessel head vent valves RCS-MOV-003A,B	4	PCCV	D	N/A	~~N/A~~ 4	~~NS~~ II	
Letdown line piping upstream of and including the letdown line stop valves RCS-VLV-021	1	PCCV	A	YES	1	I	
Reactor coolant piping drain piping upstream of and including the second drain stop valve RCS-VLV-023A, B, C, D	1	PCCV	A	YES	1	I	
Cavity / reactor coolant system water level meter piping upstream of and including the stop valves RCS-VLV-024.025	1	PCCV	A	YES	1	I	
Steam generator tube side	1	PCCV	A	YES	1	I	
Steam generator shell side	2	PCCV	B	YES	2	I	
Steam generator insulation	5	PCCV	N/A	N/A	5	II	
Pressurizer safety valves RCS-SRV-120, 121, 122, 123	1	PCCV	A	YES	1	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 4 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Safety depressurization valves RCS-MOV-117A, B	1	PCCV	A	YES	1	I	
Safety depressurization valve block valves RCS-MOV-116A, B	1	PCCV	A	YES	1	I	
Depressurization valves for severe accident RCS-MOV-118, 119	1	PCCV	A	YES	1	I	
Pressurizer spray valves RCS-PCV~~451A, B~~ -061A,B	1	PCCV	A	YES	1	I	
Pressurizer spray block valves RCS-MOV-111A, B	1	PCCV	A	YES	1	I	
(Deleted)							
Letdown line stop valve RCS-VLV-021	1	PCCV	A	YES	1	I	
Reactor coolant piping first drain stop valvesRCS-VLV-022A, B, C, D	1	PCCV	A	YES	1	I	
(Deleted)							
(Deleted)							
Pressurizer spray bypass valves RCS-VLV-112A,B	1	PCCV	A	YES	1	I	
Reactor coolant piping	1	PCCV	A	YES	1	I	
Main coolant piping insulation	5	PCCV	N/A	N/A	5	II	
Pressurizer surge line piping	1	PCCV	A	YES	1	I	
Pressurizer spray line piping	1	PCCV	A	YES	1	I	
Pressurizer relief tank	4	PCCV	D	N/A	4	II	
Reactor coolant system piping and valves related to pressurizer relief tank excluding containment isolation valves and piping between valves	4	PCCV R/B	D	N/A	4	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 9 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards[3]	Seismic Category[4]	Notes
Piping and valves contained within the demineralizer subsystem of chemical and volume control system. This subsystem branches off the letdown line between the two three way valves CVS-TCV-014 and CVS-LCV-031A. It includes reactor coolant purification filters and demineralizers, and deborating demineralizers	8	A/B	D	N/A	4	NS	
Letdown line piping and valves outside containment from the three way valve CVS-LCV-031A to the volume control tank	8	R/B A/B	D	N/A	4	NS	
Letdown line piping and valves outside containment from CVS-VLV-102, VLV-103 (including the valves) to the volume control tank	4	R/B	D	N/A	4	II	
Reactor coolant pump seal water return piping and valves outside containment from containment isolation valve CVS-MOV-204 to all other connections in chemical and volume control system	8	R/B	D	N/A	4	NS	
Reactor coolant pump seal water return piping and valves from reactor coolant pump seal to and including 4 valves CVS-AOV-192A,B,C,D	2	PCCV	B	YES	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 10 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Reactor coolant pump seal water return line drain piping and valves for the strainer and the seal water heat exchanger including the 2 valves CVS-VLV-678 and CVS-VLV-681.	8	R/B	D	N/A	4	NS	Reactor coolant pump seal water return line drain piping and valves for the strainer and the seal water heat exchanger including the 2 valves CVS-VLV-678 and CVS-VLV-681.
Reactor coolant pump seal water injection piping and valves excluding containment isolation valves, piping between these valves, piping downstream of CVS-VLV-180A, B, C, D (~~including~~ excluding valves), and seal water injection filter line valves and piping between and ~~excluding~~ including CVS-VLV-168 and CVS-VLV-173	3	R/B PCCV	C	YES	3	I	
Reactor coolant pump seal water injection piping and valves downstream of including valves CVS-VLV-180A, B, C, D	1	PCCV	A	YES	1	I	
Seal water injection filter line valves and piping between and excluding CVS-VLV-168 and CVS-VLV-173	8	R/B A/B	D	N/A	4	NS	
Charging lines from and including valves CVS-VLV-158 and CVS-AOV-159 to their penetration into the reactor coolant system	1	PCCV	A	YES	1	I	
Auxiliary spray line from and including valves CVS-AOV-155 to the penetration into the RCS	1	PCCV	A	YES	1	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 13 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards[3]	Seismic Category[4]	Notes
Chemical and volume control system piping and valves related to the boric acid pump minimum flow piping up to boric acid tank, through transfer pump minimum flow orifice and valve CVS-VLV-531A,B	4	A/B	D	N/A	4	NS	
Chemical and volume control system piping and valves related to the boric acid tanks up to and including valves CVS-SRV-509A,B, CVS-VLV-511A,B, CVS-VLV-508A,B	4	A/B	D	N/A	4	NS	
Chemical and volume control system piping and valves related to the boric acid tanks excluding foregoing piping and valves	8	A/B	D	N/A	4	NS	
Chemical and volume control system piping and valves related to the holdup tanks and the boric acid evaporator feed pumps	8	A/B	D	N/A	4	NS	
Chemical and volume control system piping and valves related to the boric acid evaporator and the boric acid evaporator feed demineralizer.	8	A/B	D	N/A	4	NS	
Chemical and volume control system piping and valves related to the primary makeup water supply isolation CVS-FCV-133A, 129, 128 and CVS-VLV-581	3	R/B	C	YES	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 16 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Residual heat removal discharge piping and valves on the reactor coolant system side between the cold legs, up to and including the second check valves RHS-VLV-027A,B,C,D	1	PCCV	A	YES	1	I	
Residual heat removal system piping and valves on the residual heat removal system side from and excluding the second motor operated valves RHS-MOV-002A, B, C, D to and excluding the second check valves RHS-VLV-027A,B,C,D	2	PCCV R/B	B	YES	2	I	
Residual heat removal system piping and valves not mentioned above up to and including the valves interfacing with systems of a lower classification.	2	PCCV R/B	B	YES	2	I	
6. Emergency Feedwater System (EFWS)							
Emergency feedwater pumps	3	R/B	C	YES	3	I	
Emergency feedwater pits	3	R/B	C	YES	5	I	
Emergency feedwater pump turbine steam drain pots	4	R/B	D	N/A	4	II~~NS~~	
Emergency feedwater pump discharge piping and valves up to and excluding emergency feedwater isolation valves EFS-MOV-019A,B,C,D	3	R/B	C	YES	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 17 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Emergency feedwater pump suction piping and valves from Emergency feedwater pits	3	R/B	C	YES	3	I	
Emergency feedwater system containment isolation valves EFS-MOV-101A,B,C,D, EFS-MOV-019A,B,C,D	2	R/B	B	YES	2	I	
(Deleted)							
Emergency feedwater pump miniflow and fullflow piping and valves to emergency feedwater pit	3	R/B	C	YES	3	I	
Emergency feedwater pump discharge tie line piping and valves	3	R/B	C	YES	3	I	
Emergency feedwater pit water supply piping and valves from Emergency feedwater pit up to and including EFS-VLV-001A,B	3	R/B	C	YES	3	I	
Emergency feedwater pump suction piping and valves from and including the valve EFS-VLV-006A,B	3	R/B	C	YES	3	I	
Emergency feedwater pump suction line piping from and including EFS-VLV-004	3	R/B	C	YES	3	I	
Emergency feedwater pit sampling piping up to and including EFS-VLV-041A,B	3	R/B	C	YES	3	I	
Emergency feedwater pit overflow piping	4	R/B	D	N/A	5	NS~~II~~	
ergency feedwater pit drain piping and valves up to and including EFS-VLV-042 A, B	3	R/B	C	YES	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 18 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Water supply piping and valves from emergency feedwater pits to spent fuel pit up to and including EFS-VLV-031	3	R/B	C	YES	3	I	
Emergency feedwater supply piping to spent fuel pit between and excluding EFS-VLV-031 and SFS-VLV-023	4	R/B	D	N/A	5	NS	
Turbine driven emergency feedwater pump steam supply piping and valves from and excluding EFS-MOV-101A,B,C,D to the pumps	3	R/B	C	YES	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves up to and including EFS-VLV-109A,B,C,D	3	R/B	C	YES	3	I	
Turbine driven emergency feedwater pump steam supply piping warming piping and valves	3	R/B	C	YES	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves up to and including EFS-VLV-117A,B, 114A,B, 111A,B, LCV-076, 077, 078, 086, 087, 088	3	R/B	C	YES	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves downstream and excluding EFS-VLV-117A,B,114A,B, 111A,B, LCV-076, 077, 078, 086, 087, 088	4	R/B	D	N/A	4	II~~NS~~	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 19 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Turbine driven emergency feedwater pump steam exhaust piping	4	R/B	D	N/A	4	II~~NS~~	
Turbine driven emergency feedwater pump steam exhaust piping drain piping	4	R/B	D	N/A	4	II~~NS~~	
Emergency feedwater pump turbine steam drain pot drain piping and valves	4	R/B	D	N/A	4	II~~NS~~	
Emergency feedwater pump turbine steam drain pot cooling water supply piping and valves	9	R/B	D	N/A	5	NS	
7. Condensate and Feedwater System							
The piping upstream of the main feedwater isolation valves ~~NFS~~FWS-SMV-512A, B, C, D to the first piping restraint at the interface between the R/B and T/B	3	R/B	C	YES	3	I	
Main feedwater piping and valves to the steam generators from and including the main feedwater isolation valves ~~NFS-VLV~~FWS-SMV-512A, B, C, D	2	R/B PCCV	B	YES	2	I	
Main feedwater piping upstream of the restraint at the interface between the R/B and the T/B	8	T/B	D	N/A	4	NS	
Emergency feedwater piping from and excluding EFS-MOV-019A,B,C,D	2	R/B	B	YES	2	I	
High pressure cleanup piping and valves in the R/B	3	R/B	C	YES	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 21 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards[(3)]	Seismic Category[(4)]	Notes
8. Main Steam Supply System (MSS)							
Main steam piping and valves including branch pipe from steam generators up to and including the following valves: 4 Nitrogen supply piping valves MSS-VLV-531A, B, C, D Main steam isolation valves MSS-SMV-515A, B, C, D Main steam bypass isolation valves MSS-HCV-565, 575, 3635, 3645 Main steam relief valves MSS-PCV-515, 525, 535, 545 Main steam depressurization valves MSS-MOV-508A,B,C,D Main steam safety valves MSS-~~VLV~~SRV-509A,B,C,D, 510A,B,C,D , 511A,B,C,D, 512A,B,C,D,513A,B,C,D, 514A,B,C,D Main steam drain isolation valves MSS-MOV-701A,B,C,D	2	R/B PCCV	B	YES	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 23 of 58)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Turbine driven emergency feedwater pump steam supply piping drain piping and valves downstream and excluding EFS-VLV-109A,B,C,D	3	R/B	C	YES	3	I	
Main steam piping from the restraint at the interface between the R/B and the T/B to the turbine	8	T/B	D	N/A	4	NS	
Main steam equalization piping	8	T/B	D	N/A	4	NS	
Main steam drain piping and valves in the turbine building	8	T/B	D	N/A	4	NS	
Nitrogen supply piping and valves up to and excluding NMS-VLV-531A,B,C,D	10	T/B	N/A	N/A	5	NS	
Turbine bypass valves MSS-TCV-550A,B,C,D,E,F,G,H,J,K,L,M,N,P,Q	8	T/B	D	N/A	4	NS	
9. Containment Spray System (CSS)							
Spray nozzles	2	PCCV	B	YES	2	I	
Containment spray system piping and valves	2	PCCV R/B	B	YES	2	I	
10. Post Accident pH Control System(PHS)							
NaTB baskets	2	PCCV	B	YES	5	I	
NaTB basket containers	2	PCCV	B	YES	2	I	
NaTB solution transfer piping	2	PCCV	B	YES	2	I	
11. Component Cooling Water System (CCWS)							
Component cooling water pumps	3	R/B	C	YES	3	I	
Component cooling water surge tanks	3	R/B	C	YES	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 39 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards[3]	Seismic Category[4]	Notes
Pressurizer liquid sampling piping and valves from hot leg up to and including the outermost containment isolation valve	2	PCCV R/B	B	YES	2	I	
Containment isolation valves PSS-MOV-071and 072 and piping between them	2	PCCV R/B	B	YES	2	I	
RHS loop sampling piping and valves up to and including the valves PSS-MOV-052A,B,C,D	2	R/B	B	YES	2	I	
Containment vessel atmosphere gas sample cooler	4	R/B	D	N/A	4	NS	
Containment vessel atmosphere gas sample moisture separator	4	R/B	D	N/A	4	NS	
Containment vessel atmosphere gas sample cooler-component cooling water side	3	R/B	C	YES	3	I	
Containment vessel atmosphere gas sampling hood	4	R/B	D	N/A	4	NS	
Containment vessel atmosphere gas sampling compressor	4	R/B	D	N/A	4	NS	
Containment vessel atmosphere sampling inlet, outlet valve PSS-MOV-301, 312	4	R/B	D	N/A	4	I	
Process and post-accident sampling systems piping and valves not specifically described above (excluding PSS-MOV-301, 312)	4	R/B A/B,AC/B	D	N/A	4	NS	
PSS-MOV-301	4	R/B	D	N/A	4	I	
PSS-MOV-312	4	R/B	D	N/A	4	I	
Sample hood and sample panel	4	AC/B	D	N/A	4	NS	
Post accident liquid sample hood	4	R/B	D	N/A	4	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 40 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Drain piping, valves in radiological controlled area	6	R/B A/B AC/B	N/A	N/A	6	Note 1	
Drain piping, valves, reactor building non-radioactive sump and sump pump in reactor building except for RCA	8	R/B	D	N/A	4	NS	
Drain piping, valves, turbine building sump and sump pump in turbine building	8	T/B	D	N/A	4	NS	
Drain piping, valves in auxiliary building and access control building, except for RCA	10	A/B,AC/B	N/A	N/A	5	NS	
Drain piping, valves in power source building	10	PS/B	N/A	N/A	5	NS	
Drain piping valves related to ESF rooms drain isolation FDS-VLV-001A,B,C,D	3	R/B	C	YES	3	I	
26. Potable and Sanitary Water System							
Potable and Sanitary Water System components, piping and valves	10	R/B,A/B,AC/B ~~PS/B,~~ T/B	N/A	N/A	5	NS	
27. Emergency Gas Turbine Auxiliary System							
Fuel oil storage tanks	3	PSFSV	C	YES	3	I	
Fuel oil transfer pumps	3	PSFSV	C	YES	3	I	
Fuel oil day tanks	3	PS/B	C	YES	3	I	
Air receivers	3	PS/B	C	YES	3	I	
Main oil pumps	3	PS/B	C	YES	5	I	
Oil cooler	3	PS/B	C	YES	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 47 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Class 1E electrical room return air fans	3	R/B	C	YES	5	I	
Class 1E battery room exhaust fans	3	PS/B	C	YES	5	I	
Dampers	3	R/B PS/B	C	YES	5	I	
Ductwork	3	R/B PS/B	C	YES	5	I	
Duct heaters	3~~5~~	R/B PS/B	C~~N/A~~	YES~~N/A~~	5	I ~~II~~	
(Deleted)							
38. Safeguard Component Area Heating, Ventilation, and Air Conditioning System							
Safeguard component area air handling units	3	R/B	C	YES	5	I	
Safeguard component area air handling unit fans	3	R/B	C	YES	5	I	
Safeguard component area air handling unit cooling coils	3	R/B	C	YES	5	I	
Safeguard component area air handling unit electric heating coils	3	R/B	C	YES	5	I	
Dampers	3	R/B	C	YES	5	I	
Ductwork	3	R/B	C	YES	5	I	
39. Emergency Feedwater Pump Area Heating, Ventilation, and Air Conditioning System							
Emergency feedwater pump area air handling units	3	R/B	C	YES	5	I	
Emergency feedwater pump area air handling unit fans	3	R/B	C	YES	5	I	

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The COL Applicant is responsible to assess the orientation of the T/G of this and other unit(s) at multi-unit site for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2.

3.5.1.3.2 Evaluation

Protection against damage from turbine missiles to safety-related SSCs is provided by the orientation of the T/G, by the robust turbine rotors, and by the redundant and fail-safe turbine design control system as described in Section 10.2. The rotor design, material selection, preservice and inservice programs and redundant control system support a very low probability of turbine missile generation. The turbine rotor design is discussed in Subsection 10.2.3, in which material selection, fracture toughness/fracture analysis is discussed. Description of the inservice inspection and testing program that will be used to maintain an acceptably low probability of missile generation is also given in Subsection 10.2.3.

The probability of unacceptable damage resulting from turbine missiles, P_4, is expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, P_1; (b) the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs, P_2; and (c) the probability of struck SSCs failing to perform their safety function, P_3.

Mathematically, $P_4 = P_1 \times P_2 \times P_3$ where RG 1.115 (Reference 3.5-6) considers an acceptable risk rate for P_4 as less than 10^{-7} per year. For ~~favorably oriented T/Gs as outlined in~~ the geometry of ~~Section~~ Subsection 3.5.1.3, ~~the product of P_2 and P_3 is conservatively estimated as 10^{-3} per year~~ the product of P_2 and P_3 is estimated as 10^{-2} per year, which is a more conservative estimate than for a favorably oriented single unit and in conformance with the guidance in SRP Section 3.5.1.3. This conservative estimation provides the flexibility for the orientation of site-specific SSCs of concern based on the guidance of RG 1.117 (Reference 3.5-19) and RG 1.115 (Reference 3.5-6). The determination of P_1 (probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing) is strongly influenced by the program for periodic inservice testing and inspection. Criteria as described in NUREG-0800 Standard Review Plan 3.5.1.3, Table 3.5.1.3-1 (Reference 3.5-7) correlates P_1 to operating cases necessary to obtain P_4 in an acceptable risk rate of 10^{-7} per year, where P_1 is less than $P_4 / (P_2 \times P_3)$ or 10^{-4}. The P_1 applicable to the US-APWR is described in Subsection 10.2.2. The COL Applicant is to commit to actions to maintain P_1 within this acceptable limit as outlined in RG 1.115, “Protection Against Low-Trajectory Turbine Missiles” (Reference 3.5-6) and SRP Section 3.5.1.3, “Turbine Missiles” (Reference 3.5-7). Reports MUAP-07028-NP, “Probability of Missile Generation From Low Pressure Turbines” (Reference 3.5–17) and MUAP-07029-NP, “Probabilistic Evaluation of Turbine Valve Test Frequency” (Reference 3.5-18) are to be used to establish programs and criteria for preservice inspection, inservice inspection interval and turbine valve test frequency in order to maintain the turbine missile generation probability, P_1, ~~equal or~~ less than the acceptable limit of 1×10^{-5} per year. Inservice inspection programs are to be maintained as outlined in SRP 3.5.1.3, Section II, Acceptance Criteria, Section 4 (Reference 3.5-7).

break. If the effects of breaks of moderate-energy fluid system piping are more severe than those of high-energy fluid system piping, then the provision of this Subsection 3.6.2.1.2.2 is applied.

Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for about 2% of the operational period but qualify as moderate-energy fluid systems for the major operational period.

3.6.2.1.2.1 Moderate-Energy Fluid System Piping in PCCV Penetration Areas

Leakage cracks are not postulated in those portion of the piping from PCCV wall to and including the inboard and outboard isolation valves provided that the PCCV penetration meets the requirements of ASME Code, Section III (Reference 3.6-10), Subarticle NE-1120 and the piping is designed so that the maximum stress range based on the sum of Equations (9) and (10) in Subarticle NC/ND-3653 of the ASME Code, Section III (Reference 3.6-9) does not exceed 0.4 times the sum of the stress limits given in NC/ND-3653.

3.6.2.1.2.2 Moderate-Energy Fluid System Piping in Areas Other than PCCV Penetrations

Leakage cracks are postulated in the following piping systems located adjacent to SSCs important to safety.

- For ASME Code, Section III, Class 1 piping, where the stress range calculated by Eq. (10) in NB-3653 is ~~less~~more than 1.2 S(m)
- For ASME Code, Section III (Reference 3.6-9), Class 2 and 3 and non-safety class piping, at axial locations where calculated stress by the sum of Equations 9 and 10 in NC/ND-3653 exceed 0.4 times the sum of the stress limits given in NC/ND-3653.
- For non-safety class piping, which has not been evaluated to obtain stress information, leakage cracks are postulated at axial locations that produce the most severe environmental effects.

3.6.2.1.3 Types of Break/Cracks Postulated

3.6.2.1.3.1 Circumferential Pipe Breaks

Circumferential breaks are postulated in high-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch at locations identified by the criteria in Subsection 3.6.2.1.1.2

No breaks are postulated in piping having a nominal diameter less than 1 inch, including instrument lines that are designed in accordance with RG 1.11 (Reference 3.6-13).

If the maximum stress range exceeds the limits specified in Subsection 3.6.2.1.1.2 and the circumferential stress range is greater than 1.5 times the axial stress range, no circumferential break is postulated; only a longitudinal break (Subsection 3.6.2.1.3.2) is postulated.

Leakage cracks are not postulated in 1-inch nominal diameter and smaller piping.

Leakage cracks are postulated in those circumferential directions that result in the most severe environmental, spray wetting, and flooding consequences.

Fluid flow from leakage cracks is based on a circular orifice with a cross-sectional area equal to that of a rectangle one-half the pipe inside diameter in length and one-half the pipe wall thickness in width. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments based on conservatively estimated time period to effect corrective actions.

3.6.2.2 Guard Pipe Assembly Design Criteria

Piping penetrations are an integral part of the PCCV pressure boundary. The annular space of the US-APWR consists of multiple compartments encircling the PCCV. These compartments segregate the PCCV electrical and mechanical penetrations into their own isolated compartments; specifically, electrical penetration rooms and mechanical penetration rooms. By virtue of the plant configuration, as piping crosses from inside to outside the PCCV, it emerges into piping penetration compartments. These compartments are designed to address postulated piping failures and the effect there of, as such, guard pipe assemblies are not required.

3.6.2.3 Analytic Methods to Define Forcing Functions and Response Models

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. To determine the forcing function for breaks postulated based on the criteria in Subsection 3.6.2.1, the fluid conditions at the upstream source and at the break exit determine the analytical approach. For most applications, one of the following situations exists.

- Superheated or saturated steam
- Saturated or sub-cooled water
- Cold water (non-flashing)

The analytical methods used for the calculation of the jet thrust for the above described situations are based on SRP 3.6.2 (Reference 3.6-3) and MHI original methodologies (Reference 3.6-25) based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31 ~~ANSI/ANS 58.2-1988 (Reference 3.6-14)~~.

The time dependent forcing function is effected by the thrust pulse resulting from the sudden pressure drop at the initial moment of pipe rupture, the thrust transient resulting from wave propagation and reflection, and the blowdown thrust resulting from the buildup of the discharge flow rate, which may reach a steady state if there is fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval.

Alternatively, a steady state jet thrust function may be used as outlined in Subsection 3.6.2.3.1.

A rise time of one millisecond is used for the initial pulse.

streamline force node orientation in the system. The flow areas and projection coefficients are described along the three axes of the global coordinate system. Each node is described by one or two flow apertures as a separate control volume. Forces are broken down orthogonally into x, y, and z components. The summation of the total number of apertures results in orthogonal thrust forces F_x, F_y, and F_z. These thrust forces are applied as input in dynamic analyses of piping and restraints.

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability

Time dependent and steady state thrust reaction loads caused by saturated or superheated steam, saturated or sub-cooled water, and cold water (non-flashing) fluid from a ruptured pipe are used in the analyses of dynamic effects of pipe breaks.

3.6.2.4.1 Jet Impingement Loading on Safety-Related Components

Structural integrity of safety-related SSCs against jet impingement load caused by pipe break is evaluated based on steady state jet force from Subsection 3.6.2.3.

Jet impingement loading is a suddenly applied constant load which can have significant energy content. These loads are generally treated as statically applied loads. The Jet impingement pressure essentially has non-uniform distributions, which varies with distance from the pipe break as shown in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31. However, the maximum pressure in the non-uniform distribution is conservatively used as a uniform distribution.

The MHI original methodologies (Reference 3.6-25) ~~methods~~ used to evaluate the jet effects resulting from the postulated breaks in high energy piping are based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31~~described in Appendices C and D of ANSI/ANS 58.2 (Reference 3.6-14)~~. Figure 3.6-2 depicts jet characteristics for the three fluid states. The short term response evaluates the jet impingement load considering a dynamic load factor of 2 and snubber supports to be active. No dynamic load factor is used and the snubbers are considered inactive for the long-term response.

3.6.2.4.1.1 Blast Wave Assessing Procedure

Computational Fluid Dynamics analysis confirms the generation of a blast wave from a steam pipe break. Potential effects are assessed on equipment within the US-APWR pressurizer compartment. Distance between the postulated pipe break locations and components is long enough to attenuate the effects. However, if layout in the pressurizer compartment is changed in the future, reassessment of the blast wave will be conducted.

Blast wave is not considered to occur from a sub-cooled water pipe break. This is because velocity of the two-phase flow at break point is slower than the speed of sound in atmospheric environments.

Therefore, the blast wave does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing a blast wave from a steam pipe break.

3.6.2.4.1.2 Jet Pressure Oscillation Assessing Procedure

Jet pressure oscillation from a steam pipe break is unlikely to occur in the US-APWR due to its high compression ratio. The jet flow expansion and Mach Disk is large. This leads to a stable downstream after the Mach Disk. The flow is so stable that disturbance at the impingement wall does not reach back to the Mach Disk.

When sub-cooled jet-flow impinges on the wall, pressure distributions on the wall are not of the concave type and a re-circulation vortex is not generated. It is because flow velocity at the jet boundary is lower than that of the core region.

Therefore, jet pressure oscillation does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing jet pressure oscillation from a steam pipe break.

3.6.2.4.1.3 Jet Reflection Assessing Procedure

When jet flow impinges on a perpendicular wall, impinged jet flow is redirected and runs along the surface of the wall. Zone of influence obtained by computational fluid dynamics is enveloped by the estimated zone of influence from MHI original methodologies (Reference 3.6-25). Inside the zone of influence, impingement pressure includes effects of pressure due to flow parallel to an impingement wall. Loads due to jet impingement reflection outside of the zone of influence are considered so small that it is not necessary to be considered.

Therefore, jet reflection does not have an impact on the design. Refer to Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing jet reflection.

3.6.2.4.2 Dynamic Analysis for Piping Systems

3.6.2.4.2.1 RCL Piping

Appendix 3C provides analysis details for RCL piping. Loads generated by postulated breaks from branch lines are applied to determine structural response of RCL piping.

3.6.2.4.2.2 Piping Other Than RCL Piping

In evaluating the dynamic effects of breaks in high-energy-fluid system piping other than RCL piping, possible break locations and break configurations are first established based on Subsection 3.6.2.1 and the effects of pipe whipping are then evaluated based on Subsection 3.6.2.4.5.

If the above evaluation determines that no safety-related SSCs are damaged, then dynamic analysis is not necessary. If the above evaluation determines that the structural integrity of safety-related SSCs is impaired, pipe whip restraints are incorporated in the high-energy-fluid system piping of concern and dynamic analysis is conducted for the system including the piping and the pipe whip restraints.

expected range of impact energies demonstrate the capability to withstand the impact without rupture. Effects on environment and shutdown logics associated with the failure of the impacted pipe are considered.

3.6.2.5 Implementation of Criteria Dealing with Special Features

Special features such as pipe whip restraints, barriers, and shields are discussed in Subsection 3.6.2.4.4.

3.6.2.6 Outline of Pipe Break Hazard Analysis Report(s)

The following information provides an outline of methodology for the pipe break hazard analysis that will be completed for high and moderate energy piping systems (including the non-safety class piping) identified in Table 2.3-1 for the closure of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) Tier 1, Table 2.3-2 related to the pipe break hazard analysis report:

- Identification of pipe break locations in high energy piping[1]
- Identification of leakage crack locations in high and moderate energy piping
- Identification of SSCs that are safety-related or required for safe shutdown[2]
- Evaluation of consequences of pipe whip and jet impingement
- Evaluation of consequences of spray wetting, flooding, environmental conditions
- Design and location of protective barriers, restraints, and enclosures

Notes

1. Table 3.6-2 provides list of high energy lines for pipe break hazard analysis, including properties of internal and external fluids.

2. All the SSCs that are safety-related or required for safe shutdown in close proximity to the postulated pipe rupture will be identified.

3.6.3 LBB Evaluation Procedures

This subsection describes the design basis to eliminate the dynamic effects of pipe rupture (Subsection 3.6.2) for the selected high-energy piping systems of RCL piping, RCL branch piping, and main steam piping. GDC 4 of Appendix A to 10 CFR 50 (Reference 3.6-1) allows exclusion of dynamic effects associated with pipe rupture from the design basis, when analyses demonstrate that the probability of pipe rupture is extremely low for the applied loading resulting from normal conditions, anticipated transients and a postulated SSE. The LBB evaluation is performed in accordance with SRP 3.6.3 (Reference 3.6-4).

The LBB analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated pipe break. The critical crack size is compared to the size of a leakage crack for which detection is certain. If the leakage crack size is sufficiently smaller than the critical crack size, the LBB requirements are satisfied.

3.6-20 Report of the ASCE Committee on Impactive and Impulsive Loads. Second ASCE Conference on Civil Engineering and Nuclear Power, Volume V, 1980.

3.6-21 Reactor Coolant Pressure Boundary Leakage Detection Systems. Regulatory Guide 1.45, U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

3.6-22 Control of the Use of Sensitized Stainless Steel. Regulatory Guide 1.44, U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

3.6-23 Evaluation of Potential Pipe Breaks, NUREG-1061, Vol. 3, U.S. Nuclear Regulatory Commission Piping Review Committee, November 1984.

3.6-24 US-APWR Leak-Before-Break Evaluation. MHI Technical Report, Later.

3.6-25 MUAP-10017, Revision 1, US-APWR Methodology of Pipe Break Hazard Analysis, December, 2010.

3.6-26 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Study of Pipe Reaction Force and Jet Impingement Load at the Pipe Break, Trans. 5th Int. Conf. on SMiRT, F6/2, 1979.

3.6-27 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Studies on Transient Water-Steam Impinging Jet, Vol. 22 No. 5, pp. 403-409, Journal of Atomic Energy Society of Japan, 1980 (in Japanese).

3.6-28 Kitade, K., Nakatogawa, T., Nishikawa, H., Kawanishi, K., and Tsuruto, C., Experimental Studies on Steam Free Jet and Impinging Jet, Vol. 22 No. 9, pp. 634-640, Journal of Atomic Energy Society of Japan, 1980 (in Japanese).

3.6-29 Masuda, F., Nakatogawa, T., Kawanishi, K. and Isono, M., Experimental Study on an Impingement High-Pressure Steam Jet, Nuclear Engineering and Design 67-2, pgs 273-285, 1982.

3.6-30 Masuda, F., Nakatogawa, T., Kawanishi, K. and Isono, M., Experimental Study on Jets Formed Under Discharges of High-Pressure Subcooled Water and Steam-Water Mixture, Trans. 7th Int. Conf. on SMiRT, F1/6, 1983.

3.6-31 Isozaki, T. and Miyazono, S., Experimental Study of Jet Discharging Test Results under BWR and PWR Loss of Coolant Accident Conditions, Nuclear Engineering and Design 96, 1986.

3.6-32 MUAP-10022, Revision 0, Evaluation on Jet Impingement Issues Associated with Postulated Pipe Rupture, January, 2011.

Table 3.6-1 High and Moderate Energy Fluid Systems

System	High-Energy[(1)]	Moderate-Energy[(1)]
Reactor Coolant System (RCS)	X	-
Chemical and Volume Control System (CVCS)	X	-
Safety Injection System (SIS)	X	-
Residual Heat Removal System (RHRS) [(2)]	-	X
Emergency Feedwater System (EFWS) [(2)]	-	X
Feedwater System (FWS)	X	-
Main Steam Supply System (MSS)	X	-
Containment Spray System (CSS)	-	X
Component Cooling Water System	-	X
Spent Fuel Pit Cooling and Purification System (SFPCS)	-	X
Essential Service Water System (ESWS)	-	X
Gaseous Waste Management System (GWMS)	-	X
Liquid Waste Management System (LWMS)	-	X
Solid Waste Management System (SWMS)	-	X
Sampling System (SS)	X	-
Steam Generator Blowdown System (SGBDS)	X	-
Refueling Water Storage System (RWS)	-	X
Primary Makeup Water System (PMWS)	-	X
Auxiliary Steam Supply System (ASSS)	X	-
Instrument Air System (IAS)	-	X
Fire Protection Water Supply ~~Service~~ System (FSS)	-	X
Station Service Air System (SSAS)	-	X
Chilled Water System (VCWS)	-	X

Notes

1. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig.

   Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered as moderate-energy.

   Piping systems that exceed 200°F or 275 psig for two percent or less of the time during which the system is in operation are considered moderate-energy.

2. The RHRS and EFWS lines are classified as moderate-energy based on the 2 percent rule. These lines experience high-energy conditions for less than 2 percent of the system operation time. The portions of the RHR system from the connections to the RCS to the first closed valve in each line are high-energy.

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 1 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pressure (psig)	Inside Pipe	Outside Pipe (°F , psig)
1	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	617	2235	Subcooled liquid	Air (120, 0)
1	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	617	2235	Subcooled liquid	Air (120, 0)
2	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	550.6	2235	Subcooled liquid	Air (120, 0)
3	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	550.6	2235	Subcooled liquid	
2	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	550.6	2235	Subcooled liquid	Air (120, 0)
3	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	550.6	2235	Subcooled liquid	
4	RCS	Surge Line	16"-RCS-2501R B	16	16	1.594	SA-312 TP316	653	2235	Saturated liquid	Air (120, 0)
5	RCS	Surge Line	16"-RCS-2501R A	16	16	1.594	SA-312 TP316	449	400	Saturated liquid	Air (120, 0)
6	RCS	Residual Heat Removal System (RHRS) Hot Leg Branch Line off RCS	10"-RCS-2501R A,B,C,D, Hot Leg Side	10	10.75	1.125	SA-312 TP316	617	2235	Subcooled liquid	Air (120, 0)
7	RCS	RHRS Cold Leg Branch Line off RCS	8"- RCS -2501R A,B,C,D (COLD LEG)	8	8.625	0.906	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
8	SIS	Accumulator System	14"-RCS-2501R A,B,C,D	14	14	1.406	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
9	RCS	Pressurizer Spray Line	6"-RCS-2501R B,C	6	6.625	0.719	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
10	MSS	Main Steam Line	32"-MSS-1532N A,B,C,D	32	32	1.496	SA333 Gr.6	535	907	Saturated steam	Air (130, 0)
11	CVS	Aux. Spray Line	3"-RCS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
12	CVS	Aux. Spray Line	3"-CVS-2561	3	3.5	0.438	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)
13	CVS	Charging Line	4"-CVS-2501	4	4.5	0.531	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)

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Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 2 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pressure (psig)	Inside Pipe	Outside Pipe (°F , psig)
14	CVS	Charging Line	4"-CVS-2561	4	4.5	0.531	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)
15	CVS	Charging Line	4"-CVS-2511 (Inside CV)	4	4.5	0.531	SA-312 TP304	130	2600	Subcooled liquid	Air (120, 0)
16	CVS	Charging Line	4"-CVS-2511 (Outside CV)	4	4.5	0.531	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
17	CVS	Charging Line	3"-CVS-2511	3	3.5	0.438	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
18	CVS	Charging Line	2"-CVS-25B1	2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
19	RCS	MCP Drain	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
20	CVS	Letdown Line	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
21	CVS	Letdown Line	3"-RCS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
22	CVS	Letdown Line	3"-CVS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
23	CVS	Letdown Line	3"-CVS-2561	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
24	CVS	Letdown Line	3"-CVS-0601	3	3.5	0.216	SA-312 TP304	380	350	Subcooled liquid	Air (120, 0)
25	CVS	Letdown Line	4"-CVS-0601	4	4.5	0.237	SA-312 TP304	380	350	Subcooled liquid	Air (120, 0)
26	CVS	Letdown Line	4"-CVS-06A1	4	-	-	-	200	350	Subcooled liquid	Air (105, 0)
27	SIS	Emergency Letdown Line	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)
28	SIS	DVI Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
29	SIS	SI Pump Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)
30	SIS	SI Pump Line	4"-SIS-2501	4	4.5	0.531	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 3 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pressure (psig)	Inside Pipe	Outside Pipe (°F , psig)
31	RCS	Pressurizer Safety Valve Line	6"-RCS-2501	6	6.625	0.719	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
31	RCS	Pressurizer Safety Depressurization Valve Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
32	RCS	Pressurizer Safety Depressurization Valve Line	6"-RCS-2501	6	6.625	0.719	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
33	RCS	Pressurizer Safety Depressurization Valve Line	8"-RCS-2501	8	8.625	0.906	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
34	CVS	Seal Injection Line	1-1/2"-CVS-2501	1-1/2	1.9	0.281	SA-312 TP316	130	2266	Subcooled liquid	Air (120, 0)
35	CVS	Seal Injection Line	1-1/2"-CVS-2511	1-1/2	1.9	0.281	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
36	CVS	Seal Injection Line	1-1/2"-CVS-25B1	1-1/2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
37	CVS	Seal Injection Line	1"-CVS-2511	1	1.315	0.250	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
38	CVS	Seal Injection Line	2"-CVS-2511	2	2.375	0.344	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
39	CVS	Seal Injection Line	2"-CVS-25B1	2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
40	SIS	Accumulator Tank Drain Line	2"-SIS-06A1	2	-	-	-	300	700	Subcooled liquid	Air (120, 0)
41	SIS	Accumulator Tank Line	14"-SIS-2511	14	14	1.406	SA-312 TP304	300	2485	Subcooled liquid	Air (120, 0)
42	SIS	Accumulator Tank Line	14"-SIS-0601	14	14	0.500	SA-312 TP304	300	700	Subcooled liquid	Air (120, 0)
43	EFS	Emergency Feedwater Pump Line	3"-FWS-1522	3	3.5	0.300	SA-106 Gr.B	471	1185	Subcooled liquid	Air (130, 0)
44	EFS	Emergency Feedwater Pump Turbine Line	6"-EFS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Subcooled liquid	Air (130, 0)
45	EFS	Emergency Feedwater Pump Turbine Line	6"-MSS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Subcooled liquid	Air (130, 0)
46	FWS	Feedwater Line	18"-FWS-1805	18	18	1.375	SA-335 Gr.P22	471	1850	Subcooled liquid	Air (130, 0)

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Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Inte uids (Sheet 4 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pressure (psig)	Inside Pipe	Outside Pipe (°F , psig)
47	FWS	Feedwater Line	6"-FWS-1805	6	6.625	0.562	SA-335 Gr.P22	471	1850	Subcooled liquid	Air (130, 0)
48	FWS	Feedwater Line	16"-FWS-1525	16	16	0.844	SA-335 Gr.P22	471	1185	Subcooled liquid	Air (130, 0)
49	FWS	Feedwater Line	3"-FWS-1802	3	3.5	0.438	SA-106 Gr.B	471	1850	Subcooled liquid	Air (130, 0)
50	MSS	Main Steam Line	32"-MSS-1532	32	32	1.500	SA-333 Gr.6	539	938	Saturated steam	Air (130, 0)
51	MSS	Main Steam Line	6"-MSS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Saturated steam	Air (130, 0)
52	MSS	Main Steam Drain Line	2"-MSS-1532	2	2.375	0.218	SA-106 Gr.B	539	938	Saturated liquid	Air (130, 0)
53	MSS	Main Steam Drain Line	4"-MSS-1532	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (130, 0)
54	SGS	SGBD Line	3"-SGS-1532	3	3.5	0.300	SA-106 Gr.B	539	938	Saturated liquid	Air (120, 0)
55	SGS	SGBD Line	4"-SGS-1532 (Inside CV)	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (120, 0)
56	SGS	SGBD Line	4"-SGS-1532 (Outside CV)	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (105, 0)
57	SGS	SGBD Line	3/8"-SGS-2521	3/8	-	-	-	539	938	Saturated liquid	Air (120, 0)
58	SGS	SGBD Line	3/8"-SGS-25CA	3/8	-	-	-	539	938	Saturated liquid	Air (105, 0)

The damping values for systems that include two or more substructures, such as a concrete and steel composite structure, can be obtained using the strain energy method. The strain energy dependent modal damping values are computed based on Reference 3.7-18, which is the same as the stiffness weighted composite modal damping method, and acceptable to SRP 3.7.2 (Reference 3.7-16).

The stiffness weighted modal damping ratio h_i of the j^{th} mode is obtained from the following equation:

$$h_j = \frac{\vec{\phi}_j^{\,T}\left[\overline{K}\right]\vec{\phi}_j}{\vec{\phi}_j^{\,T}\left[K\right]\vec{\phi}_j}$$

where

$[K]$	= the stiffness matrix of the combined soil-structure system
$\vec{\phi}_j$	= the j^{th} normalized mode shape vector
$\left[\overline{K}\right] = \sum [k_i] \cdot \xi_i$	= the modified stiffness matrix constructed from the products of the element stiffness matrices $[k_i]$ and the applicable damping ratio ξ_i

Formulation of damping values for the seismic analysis models which incorporate the combined soil-structure damping is discussed in Subsection 3.7.2.1. Damping values associated with site-specific SSI analyses are addressed in Subsection 3.7.2.4.1.

3.7.1.3 Supporting Media for Seismic Category I Structures

A range of soil parameters of the basemat supporting media are considered in the seismic design of seismic category I building structures for the US-APWR standard plant. The overall basemat dimensions, basemat embedment depths, and maximum height of the US-APWR R/B, PCCV, and containment internal structure on their common basemat are given in Table 3.7.1-3 and as updated by the COL Applicant to include site-specific seismic category I structures.

The required allowable static bearing capacity for seismic category I building structure basemats, including the R/B-PCCV-containment internal structure on their common basemat, is 15 ksf. The dynamic bearing loads for seismic category I structure basemats are dependent upon the magnitude of the seismic loads that can be obtained from a site-specific seismic analysis that considers FIRS. The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, and to evaluate the bearing load to this capacity. A minimum factor of safety of 2.5 is suggested for the ultimate bearing capacity versus the allowable static bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions. A minimum factor of safety of 2~~.5~~ is suggested for the ultimate bearing capacity versus the allowable dynamic bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions.

The site-independent seismic design of seismic category I and seismic category II SSCs uses lumped parameter representation to model the interaction of seismic category I structures with the supporting media. The lumped parameter model considers a rigid basemat resting on the surface of a uniform elastic-half-space. Six sets of two

The site-specific SSI analyses will use site-dependent input control motion that is derived from GMRS and FIRS discussed in Subsection 3.7.1.1. The primary non-linear material behavior of the soil must be considered and may be approximated by using equivalent linear material properties that are compatible to the free-field strains generated by the site-specific design ground motion. If the earthquake-induced strains in the soil remain below 2%, the strain-compatible soil properties are obtained from a 1-dimensional wave propagation analysis by using equivalent-linear methodology and site-specific soil stiffness and damping degradation curves. The site-specific SSI analyses of the R/B–PCCV-containment internal structure on their common basemat uses the finite element (FE) analysis program ACS SASSI ~~Version 2.2~~ (Reference 3.7-17) that provides a frequency domain solution of the SSI model response by using a sub-structuring technique and, when applicable, is capable of addressing site-specific effects such as the layering of the soil, embedment and flexibility of the basemat, scattering, and incoherence of the input control motion. Based on successful comparison of ISRS derived from the CSDRS to those derived from site-specific SASSI analysis, other standard plant structures designed using lumped parameter models with lumped SSI parameters subject to the CSDRS can be validated by direct comparison to demonstrate their site-specific FIRS are enveloped. A SASSI analysis can be performed to consider incoherency to reduce high frequency response.

The lumped parameter model and the input soil properties used for the site-independent SSI analysis of the US-APWR standard plant, as well as the suggested methodologies for analyzing the effect of site-specific conditions on SSI response, are discussed further in Subsection 3.7.2.4.

3.7.2 Seismic System Analysis

Seismic system analysis is discussed in the following Subsections, 3.7.2.1 through 3.7.2.15. Following the guidance of the acceptance criteria section II.3(a) of SRP 3.7.2 (Reference 3.7-16), two categories of seismic category I SSCs are defined: (1) seismic systems that include major seismic category I buildings and structures that are analyzed in conjunction with their basemats and supporting media (subgrade); and (2) seismic subsystems that include other seismic category I SSCs that are not analyzed in conjunction with basemats and subgrade. This subsection discusses the following major seismic category I and II buildings and structures that are classified as seismic systems requiring SSI analysis:

- R/B–PCCV-containment internal structure on their common basemat (seismic category I)
- East and west PS/Bs (seismic category I)
- A/B (seismic category II)

The seismic responses of the major seismic category I and seismic category II structures are required to be obtained from frequency domain time history analysis of seismic models considering a frequency-dependent SSI system, including a set of eight generic layered soil profiles representing a wide range of site conditions. Subsections 3.7.2.1 and 3.7.2.3, respectively, describe the analysis and modeling methods used for the seismic analyses, and Subsection 3.7.2.2 discusses the natural frequencies and results obtained from the seismic analyses. The results from the seismic analyses serve as the basis for the development of equivalent static seismic loads that are applied in conjunction with other design loads on the detailed three-dimensional shell FE model in order to obtain the design stresses in the structural members and components.

coupling of the containment internal structure with the equipment and the piping, the analysis of the R/B complex includes lumped mass stick models representing the stiffness and mass inertia properties of the major equipment and piping, including the reactor vessel and reactor coolant loop. The methodology initially used to develop the stick models and the stick model properties is presented in Technical Report MUAP-10001 ~~08005~~ (Reference 3.7-47~~18~~), and in the following Subsections 3.7.2.3.4 through 3.7.2.3.9. The methodology in Subsections 3.7.2.3.4 through 3.7.2.3.9 is enhanced by: adjusting member properties where necessary to account for the effects of concrete cracking, incorporating single degree of freedom models representing the out-of-plane flexibility of slabs and walls, and integrating the stick models with a finite element model of the R/B complex basement to form the overall SSI model of the R/B complex. Technical Report MUAP-10001 (Reference 3.7-47) presents the methodology used for these enhancements and describes the overall enhanced R/B complex model in further detail.

The lumped-mass stick models used for the seismic analysis of US-APWR R/B, PCCV, and containment internal structure and their basemat consider the eccentricities between the center of rigidity and the center of mass of structures. The models represent the actual locations of masses and centers of rigidity, thus, accounting for the torsional effects caused by the eccentricity. The modeling approach accounts for the differences between the vertical and horizontal centers of rigidity by using two stick elements to model the stiffness of the structural members at each story. A truss element located at the vertical center of rigidity represents the vertical stiffness of the floor, and a beam element located at the shear center of rigidity represents the shear and bending stiffness of the floor. Both stick elements are rigidly connected to the common center of mass at each major floor elevation. This modeling approach helps eliminate the errors in computation of the seismic responses that are due to the rocking SSI effects caused by an inaccurate representation of the vertical center of rigidity. See Subsection 3.7.2.11 for discussion of accidental torsion.

To model the interaction with the underlying subgrade, the standard plant R/B complex model is analyzed in conjunction with a set of eight generic layered profiles using the program ACS SASSI (Reference 3.7-17) as discussed in Subsection 3.7.2.4.

The structural elements of the R/B, which includes the fuel handling area, are concentrated and reduced to one set of stick models below the operating floor level. The part of the R/B above the operation floor level, except for the fuel handling area, is represented by several stick models that are interconnected by horizontal rigid links representing the floor diaphragm. The rigid links restrain only the in-plane translational displacements of the floor without affecting the deformations in the other DOF. The containment internal structure and the PCCV are modeled separately, and they are rigidly connected to the R/B stick model at the surface of the common basemat. The R/B, PCCV, and containment internal structure are all structurally separated from each other above their common basemat by expansion joints, which are discussed further in Subsection 3.7.2.8. Detailed descriptions of the R/B, PCCV, and containment internal structure and their common basemat are presented in Section 3.8, where the structural design of those buildings and structures is addressed.

A set of static and dynamic analyses is performed on the detailed three-dimensional FE model that is developed for computation of internal forces and stresses in the structural members and components of R/B-PCCV-containment internal structure subject to design loads and load combinations that are discussed further in Section 3.8. The FE

- If 0.01 less than or equal to R_m and less than or equal to 0.1, decoupling can be done if 0.8 is greater than or equal to R_f and greater than or equal to 1.25
- If R_m is greater than 0.1, a subsystem model should be included in the primary system model

where

R_m = (total mass of supported subsystem)/(total mass of supporting system)

R_f = (fundamental frequency of supported system)/(dominant frequency of support motion)

If these criteria require the subsystem to be coupled with the primary seismic model, both the stiffness and the mass of the subsystem are included in the overall model to assure the accuracy of the calculated frequencies. This is the approach used for including the RCL seismic subsystem in the coupled RCL-R/B-PCCV-containment internal structure lumped mass stick model discussed in Technical Report MUAP-10001~~08005~~ (Reference 3.7-47~~18~~).

When it has been determined through investigation of the above criteria that a subsystem is not required to be coupled with the primary seismic model, then the subsystem is assumed absolutely rigid and only its mass is included at appropriate node points of the global seismic model. The PCCV polar crane and fuel handling crane are incorporated into the overall lumped mass stick model in this manner. In addition, the requirements of NOG-1 (Reference 3.7-22) for the design of cranes may require that the crane design analysis be performed by coupling the crane model with the overall building model. If found that is required, the site-specific seismic analysis of the US-APWR standard plant must be performed on models that incorporate the PCCV polar crane and the fuel handling crane, as appropriate.

3.7.2.3.5 Section and Material Properties for Lumped Mass Stick Models

The values of the modulus of elasticity and Poisson's ratio (ν) for concrete and steel used in the lumped mass stick models are discussed below. The values are for materials at or near ambient temperatures.

(a) Concrete

The concrete modulus of elasticity E_c, and shear modulus G_c corresponding to the compressive strengths of normal weight concrete used in the R/B, PCCV, and containment internal structure and their common basemat are summarized in Table 3.7.2-2 and are computed as follows:

$$E_c \text{ (ksi)} = 57{,}000 \sqrt{f'_c}$$

$$G \text{ (ksi)} = E_c / 2\,(1 + \nu_c)$$

where

f'_c = specified 28-day compressive strength of concrete (psi)

The equivalent dead loads (mass) are appropriately increased in areas such as main piping corridors, and cable tray and HVAC ductwork runs where such loads exceed the value of 50 lb/ft^2.

3.7.2.4 Soil-Structure Interaction

In accordance with the requirements of SRP 3.7.2, Section II.4 (Reference 3.7-16), SSI effects are considered in the seismic response analysis of all major seismic category I and seismic category II buildings and structures that are part of the US-APWR standard and non-standard plant. The SSI analyses use seismic models that are described above in Section 3.7.2.3 to represent the dynamic properties of the structures. The ACS SASSI (Reference 3.7-17) PS/Bs model and enhanced R/B lumped mass stick model, combined in ACS SASSI with the FE model of the basement, provide adequate degrees of freedom to ensure that the modeling requirements of SRP 3.7.2, Section II A (iv) are met, and that the seismic response in the high frequency range is captured. For the R/B complex and PS/B~~s~~ model, site-specific SSI analyses are also performed using the computer program ACS SASSI in order to confirm that site-specific effects are enveloped by the site-independent analyses.

The site-independent SSI analyses take into account the flexibility of the basemat, frequency-dependence of the SSI impedance, layering of the subgrade, elevation of the water table, and scattering of the input motion. The SSI analyses are performed with ACS SASSI (Reference 3.7-17) in the frequency domain utilizing the substructuring technique and complex stiffness representation of stiffness and damping properties of the structures and the subgrade. The subgrade media and SSI system damping to model the dissipation of energy due to material damping of the structural members and the soil are also discussed in Subsections 3.7.1.3 and 3.7.1.2. The response of the system at selected frequencies of analyses is obtained as the solution of a set of complex algebraic equations. The frequencies of analyses are selected to accurately capture the response of the structure at all important frequency ranges. The amplitudes of the interpolated transfer functions are plotted and investigated to ensure the accuracy of the interpolation of the response for the required range of frequencies. Approaches and methods used for the SSI analyses are discussed further in Technical Report MUAP-10001 (Reference 3.7-47).

The ratio of basemat depth-to-equivalent-radius for the R/B-PCCV basemat is approximately 0.27. ASCE 4-98 Subsection 3.3.4.2 (Reference 3.7-9) considers that a basemat depth-to-equivalent-radius ratio of less than 0.3 is an indication of a shallow embedment foundation, for which effects of the embedment on the seismic response of the building are generally not significant. SSI analysis performed as part of the site-independent US-APWR standard plant design neglects the effects of embedment of the common R/B and PCCV basemat. Therefore, the R/B-PCCV seismic models are not coupled with any subgrade or backfill material at the sides of the basemat or along the faces of below-grade exterior walls, and no credit is taken in the seismic analysis for reduction in amplitude of the response due to foundation embedment in the subgrade or backfill materials. Embedment effects, including shifts in the structural frequencies, are considered to be small enough to be enveloped by the variations of subgrade stiffness considered in the standard design seismic response analyses of a surface foundation. However, the effects of the embedment are required to be analyzed on a site-specific basis as discussed in Subsection 3.7.2.4.1 to confirm suitability of design.

The use of frequency independent SSI impedance parameters is based on the assumption that the subgrade conditions are relatively uniform up to a depth of one equivalent basemat diameter below the bottom of the basemat of the major seismic category I structures. Dry soil conditions are assumed in order to simplify the analysis. The following values for shear wave velocity V_s, density γ and Poisson's ratio ν are assigned to the uniform elastic half-space to simulate the general subgrade conditions:

- Soft soil site, V_s = 1,000 ft/s, γ = 110 pcf, ν = 0.40
- Rock site (Medium 1), V_s = 3,500 ft/s, γ = 130 pcf, ν = 0.35
- Rock site (Medium 2), V_s = 6,500 ft/s, γ = 140 pcf, ν = 0.35
- Hard rock site, V_s = 8,000 ft/s, γ = 160 pcf, ν = 0.30

A fixed base analysis considers the hard rock case listed above. The values used for the soil shear wave velocities are considered to be compatible to the strain level corresponding to the site-independent SSE. Table 3.7.2-3 summarizes the US-APWR standard plant seismic SSI analysis cases, with respect to the input time histories applied to the stick models resting on the uniform elastic half-space having the different subgrade conditions listed above.

The site-specific SSI analyses take into account site-specific conditions such as soil layering, location of water table and embedment of the basemat and, thus, validate the results of the site-independent SSI analysis and assumptions contained in the US-APWR standard plant design. This is accomplished through site-specific SSI analysis as explained below.

3.7.2.4.1 Requirements for Site-Specific SSI Analysis of US-APWR Standard Plant

The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS~~-~~_SASSI ~~SSI Version 2.2~~ (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design. After the SASSI analysis is first performed for a specific unit, subsequent COLAs for other units may be able to forego SASSI analyses if the FIRS and GMRS derived for those subsequent units are much smaller than the US-APWR standard plant CSDRS, and if the subsequent unit can also provide justification through comparison of site-specific geological and seismological characteristics.

SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. Consideration of structure-to-structure interaction is discussed in Subsection 3.7.2.8. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:

- Seismic category I ESWPT
- Seismic category I PSFSV
- Seismic category I UHSRS

- The basemats are much stiffer than the supporting subgrade
- The SSI impedance functions remain relatively constant in the range of frequencies important for the design
- The consideration of basemat embedment yields conservative results

In accordance with SRP 3.7.2 (Reference 3.7-16), Section II.4, fixed base response analysis can be performed if the basemats are supported by subgrades having a shear wave velocity of 8,000 ft/s or higher, under the entire surface of the foundation.

3.7.2.5 Development of Floor Response Spectra

ISRS for the PS/Bs and RCL-R/B-PCCV-containment internal structure are developed from the results of the site-independent seismic analyses of the seismic models described in Subsection 3.7.2.3 by applying methods described in Subsection 3.7.2.1, and capturing SSI effects as described in Subsection 3.7.2.4, using generic soil profiles described in Subsection 3.7.1.3. The statistically independent time histories developed from the CSDRS as described in Subsection 3.7.1.1 serve as input control motion in the analysis. Note that the dynamic properties of the stick model portions of the R/B complex seismic model presented in Technical Report MUAP-10001 (Reference 3.7-47) are modified to account for the effects of cracking for accuracy in the seismic design and development of the ISRS. The ISRS are derived from the calculated responses at locations and elevations where major seismic category I and II SSCs are located. ~~ISRS for the major seismic category I structures as well as design spectra for the RCL system are developed from the results of the site-independent seismic analysis of the coupled RCL-R/B-PCCV-containment internal structure lumped mass stick model described in Technical Report MUAP-08005 (Reference 3.7-18) by using direct integration time history analysis method as described in Subsection 3.7.2.1, and by capturing SSI effects for all four generic soil conditions as discussed in Subsection 3.7.2.4. The statistically independent time histories developed from the CSDRS as described in Subsection 3.7.1.1 serve as input control motion for the analysis. The dynamic properties of the stick models are discussed in detail in Appendix 3H for the R/B-PCCV-containment internal structure and the east and west PS/Bs. The ISRS are derived from the calculated responses of the R/B-PCCV-containment internal structure and PS/Bs lumped mass stick models at locations and elevations where major seismic category I and II SSCs are located.~~

In developing the ISRS, the effects of floor slab system out-of-plane flexibility are considered by investigating floor slab systems (using local FE models or other means of analysis), independently from the overall lumped mass stick model in order to determine their natural frequencies. Depending on the results, the floor slab systems may then be analyzed as simple single DOF vertical oscillators to determine maximum accelerations (ZPA values) to be used for development of the ISRS for the respective floor locations. The concrete cracking of slabs is considered in the development of the single DOF models, in accordance with ACI 349-01 (Reference 3.7-31). If the results of the independent modal analyses indicate that higher modes of vibration have to be considered, the floor systems may also be analyzed as subsystems, as described further in Subsection 3.7.3.1. The local analyses of floor slab systems with respect to out-of-plane flexibility and effects on the ISRS are addressed in Technical Reports MUAP-10001 and MUAP-10006 (References 3.7-47 and 3.7-48).~~The local analyses of floor slab systems with respect to out-of-plane flexibility and effects on the ISRS are addressed as part of a later Technical Report.~~

The SSE ISRS for seismic category I buildings and structures of the US-APWR standard plant are developed directly from the results of the site independent seismic analysis. As previously explained in Subsection 3.7.1.1, since the OBE ground motion is limited to a maximum of 1/3 times the CSDRS, explicit design and analysis for OBE is not required, as permitted by 10 CFR 50 Appendix S (Reference 3.7-7). Therefore, separate OBE ISRS are not developed for design and analysis of US-APWR standard plant systems and subsystems.

In the case where seismic qualification by testing is performed in accordance with IEEE Std 344-2004~~1987~~ (Reference 3.7-13~~25~~), test response spectra which replicate the OBE response spectra are not required since the OBE condition is no longer used as a design basis. The US-APWR program for seismic and dynamic qualification of mechanical and electrical equipment is discussed in Section 3.10.

The ISRS are developed for damping values equal to 0.5%, 2%, 3%, 4%, 5%, 7%, 10%, 20% of critical damping and for variable damping where permitted by ASME Code Case N411-1, as discussed in RG 1.61 (Reference 3.7-15). The ISRS envelope the spectra obtained from the site-independent analyses for all generic subgrade conditions described in Subsection 3.7.1.3. ISRS developed from the site-independent seismic analyses of the R/B complex and PS/Bs are used for design. ISRS developed at 5% critical damping, which are presented in Technical Report MUAP-10006 (Reference 3.7-48) and referenced in Appendix 3I are used to validate the standard plant ISRS by comparison to site-specific ISRS that are also developed at 5% critical damping. The process for developing enveloped ISRS is described in detail in Section 3.5 of Technical Report MUAP-10006 and is summarized as follows: ~~The ISRS envelope the spectra obtained from the site-independent analyses for all four of the different generic subgrade conditions. Figure 3.7.2-11 outlines the development of the enveloped design ISRS, for which Figure 3.7.2-12 provides an example of a design ISRS. Design ISRS for R/B-PCCV-containment internal structure are provided in Appendix 3I. The process for developing enveloped ISRS is as follows:~~

- The response spectra are generated for the three components of earthquake by SRSS, following the general guidance of RG 1.122 (Reference 3.7-26) for frequencies up to 100 Hz.
- The maximum spectral acceleration at each frequency obtained from the seismic analysis of any general subgrade conditions is selected for the envelope.
- The enveloped ISRS are smoothed and broadened by +/-15%. The valleys in the enveloped ISRS are filled when necessary to capture potential shifts in the seismic response caused by soil properties that are different from, but bounded by, the four generic soil conditions of the standard plant. Alternatively in some locations, the peak shifting method described in Subsection 3.7.3.1 can be used instead of the broadened response spectra method.

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piping, and other plant SSCs is in accordance with IEEE Std 344-2004~~1987~~ (Reference 3.7-13~~25~~) and is addressed in Section 3.10.

3.7.3.2 Procedures Used for Analytical Modeling

Seismic subsystems are defined as those systems that are not analyzed in conjunction with basemats and subgrade, as previously discussed in Subsection 3.7.2. The procedures used for analytical modeling of subsystems include the use of mathematical computer models comprised of nodes and elements used to represent connections and members. Depending on the complexity of the subsystem, the models may be lumped mass stick models or FE models. The models contain sufficient detail and DOFs to represent the structural and seismic response of the subsystem, and are incorporated into the overall building model when required by the coupling criteria discussed in Subsection 3.7.2.3.4. Depending on the complexity of the seismic subsystem, structure, or component being analyzed, detailed member design may be performed by hand calculations using the results of the overall building structural and seismic analyses. Alternatively, the computer model may be sufficiently detailed to be used for the design calculation of the individual members. In all cases, the computer programs used for analytical modeling of subsystems are verified and validated in accordance with ANSI/ASME NQA-1-2004 (Reference 3.7-23) requirements.

3.7.3.3 Analysis Procedure for Damping

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The SSE damping values to be used in the dynamic analysis for various seismic category I and II subsystems and their related supports are shown in Table 3.7.3-1(a). The damping values are based on RG 1.61 (Reference 3.7-15). The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7% of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining NRC review for acceptance on a case-by-case basis.

For subsystems that are composed of different material types, the composite modal damping approach with either the weighted mass or stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems.

Piping systems are analyzed for SSE using 4% damping. Alternatively, frequency-dependent damping values may be utilized as noted and described in Tables 3.7.3-1(a) and 3.7.3-1(b). The seismic analysis of piping and other mechanical subsystems is addressed in further detail in Sections 3.9 and 3.12.

3.7.3.4 Three Components of Earthquake Motion

For seismic category I subsystems, the three components of earthquake motion are considered in the same manner as described in Subsection 3.7.2.6.

3.7.4.1 Comparison with Regulatory Guide 1.12

The proposed seismic instrumentation program is generally in accordance with RG 1.12 and RG 1.166 (References 3.7-40, 3.7-41), and consistent with the methodology used for seismic analysis that is discussed in Subsection 3.7.2. The seismic design of US-APWR standard plant is based on site-independent seismic response analysis of basemats resting on generic supporting media that are ~~the surface of elastic half-space that is~~ subjected to the CSDRS input ~~a~~ control motion. The site independent OBE is defined as 1/3 of the CSDRS presented in Subsection 3.7.1.1. Verification of the site-independent standard design is performed during seismic analyses that consider site-specific conditions, such as soil layering, basemat embedment, water table depth etc. The FIRS, which are developed consistent with the ~~from~~ site-specific GMRS~~,~~ define the site-specific control design motion.

The criteria that define the vibratory motion that requires the shutdown of the US-APWR plant are based on the site-specific OBE. The 5% damping FIRS associated with the site-specific OBE must be enveloped by 1/3 of the 5% damping CSDRS. The conditions that require a shutdown of the US-APWR plant are defined by the site-specific OBE ISRS at the locations of seismic instrumentation. Unless site-specific OBE is set at 1/3 of the site-specific SSE or lower, these spectra shall be obtained from SSI analysis using as input the site-specific OBE FIRS and properties of the supporting media that are strain-compatible to the site-specific OBE ground motion. When the site-specific OBE is equal or lower than 1/3 of site-specific SSE, the spectra scaled from the 5% damping site-specific SSE response spectra may be used directly for OBE exceedance checks. The measured response spectra at each of the instrumentation locations in Subsection 3.7.4.2 are compared against the corresponding site-specific OBE instructure acceleration and velocity response spectra in accordance with RG 1.166 (Reference 3.7-41). The comparison evaluation is to be performed within 4 hours of the earthquake using data obtained from the three components of the earthquake motion as defined by the three orthogonal axes of the standard plant (two horizontal and one vertical) on the uncorrected earthquake records. The evaluation is also to include a check on the operability of the seismic instrumentation as mandated by Section 4.3 of RG 1.166 (Reference 3.7-41).

The locations of seismic monitors for the US-APWR standard plant are provided in Subsection 3.7.4.2. The COL Applicant is to verify the site-specific applicability of these monitors, and determine if there is a need for the installation of additional instrumentation for the measurement of the free-field ground motion based on conditions and requirements specific to the site. The CAV is based on criteria for exceedance of OBE using measurements taken in the free-field, however the OBE exceedance can be evaluated by using only measurements from instrumentation installed on the buildings and the structures of the US-APWR standard plant. The seismic instrumentation for monitoring the free-field ground motion is not specifically required since both the site-independent and site-specific design are based on control motions that are defined at the bottom of the basemats. The response spectra of the free-field ground motion are not directly relevant to the design of the US-APWR standard plant nor are directly comparable to the design input ground motion as defined by the CSDRS and FIRS in Subsection 3.7.1.1.

The calculation of the CAV is performed in the manner provided in Electric Power Research Institute (EPRI) Report TR-100082 (Reference 3.7-42). As stated in RG 1.166 (Reference 3.7-41), the range of the spectral velocity limit should be 1.0 to 2.0 Hz which

- On level 2F of PCCV at elevation 25 ft, 3 in., located in the southwest quadrant outside the steam generator and reactor coolant compartment.
- On level 4F of PCCV operating deck slab at elevation 76 ft, 5 in., located in the southwest quadrant outside the steam generator and reactor coolant compartment underneath the access stairs adjacent to the west PCCV buttress.
- On the basemat of the east PS/B on the B1F level at elevation -23 ft, 4 in., in the non safety-related turbine generator anteroom.
- On level 1F of the east PS/B at elevation 3 ft, 7 in., in the non safety-related turbine generator control room.

The locations listed above correlate to structural elements in the structures which have been modeled as mass points in the dynamic analysis so that the measured motion can be directly compared to the design spectra. The instrumentation mounted at the locations listed above is not mounted on equipment, piping, supports, or secondary structural frame members. These locations have been reviewed in accordance with RG 8.8 (Reference 3.7-44) and determined to be consistent with maintaining dose rates as low as practical and maintaining occupational radiation exposures as low as is reasonably achievable for access and maintenance of the instrumentation.

A time-history analyzer/recorder is provided which has the capability to provide pre-event recording time of 3 seconds minimum and post-event recording time of 5 seconds minimum, and to record at least 25 minutes of sensed motion. The recorder portion of the time-history analyzer is to have the capability of a sample rate of at least 200 samples per second in each of the three orthogonal directions of the plant, a bandwidth of 0.20 Hz to 100 Hz, and a dynamic range of 1,000:1 zero to peak. The triaxial acceleration sensors are to have the same dynamic range as the time-history analyzer recorder and a frequency range of 0.20 Hz to 100 Hz. The triggers of the tri-axial acceleration sensor units are to be capable of being set within the range of 0.001g to 0.02g. Batteries are provided with enough capacity for a minimum of 25 minutes of system operation at any time over a 24-hour period, without recharging, in combination with a battery charger whose line power is connected to an uninterruptible power supply.

The seismic instrumentation serves no safety-related function and, therefore, has no nuclear safety design requirements. However, its design and location are in accordance with RG 1.12 (Reference 3.7-40), which requires that the seismic instrumentation:

- will not be affected by the failure of adjacent SSCs during an earthquake;
- will operate during all modes of plant operation, including periods of plant shutdown; and
- is protected as much as practical against accidental impacts.

As required by RG 1.12 (Reference 3.7-40), the seismic instrumentation is rigidly mounted and oriented so that the horizontal components are parallel to the horizontal axes of the standard plant used in the seismic analyses. These features of the seismic monitoring instrumentation are obtained by qualifying the equipment to IEEE Std 344-2004~~1987~~ (Reference 3.7-13~~25~~); the seismic qualification program is discussed in Section 3.10.

COL3.7(24) The COL Applicant is to verify that the site-specific ratios V/A and AD/V^2 *(A, V, D, are PGA, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.*

COL3.7(25) The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B-PCCV-containment internal structure, and PS/B model, utilizing the program ACS~~-~~ SASSI ~~SSI Version 2.2~~ (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design. After the SASSI analysis is first performed for a specific unit, subsequent COLAs for other units may be able to forego SASSI analyses if the FIRS and GMRS derived for those subsequent units are much smaller than the US-APWR standard plant CSDRS, and if the subsequent unit can also provide justification through comparison of site-specific geological and seismological characteristics.

COL3.7(26) SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. Consideration of structure-to-structure interaction is discussed in Subsection 3.7.2.8. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:

- *Seismic category I ESWPT*
- *Seismic category I PSFSV*
- *Seismic category I UHSRS*

COL3.7(27) It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.

COL3.7(28) The overall basemat dimensions, basemat embedment depths, and maximum height of the US-APWR R/B, PCCV, and containment internal structure on their common basemat are given in Table 3.7.1-3 and as updated by the COL Applicant to include site-specific seismic category I structures.

COL3.7(29) Table 3.7.2-1, as updated by the COL Applicant to include site-specific seismic category I structures, presents a summary of dynamic analysis and combination techniques including types of models and computer programs used, seismic analysis methods, and method of combination for the three directional components for the seismic analysis of the US-APWR standard plant seismic category I buildings and structures.

3.7-12 United States Nuclear Regulatory Commission Staff Requirement Memorandum SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, James M. Taylor, Executive Director of Operations, April 2, 1993.

3.7-13 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, ~~Appendix D,~~ Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005.

3.7-14 McGuire, R.K., Silva, W.J., and Costantino, C.J. Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines, NUREG/CR-6728, U.S. Nuclear Regulatory Commission, Washington, DC, October 2001.

3.7-15 Damping Values for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.61, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.7-16 Seismic System Analysis, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 3.7.2, Rev.3, United States Nuclear Regulatory Commission, March 2007.

3.7-17 An Advanced Computational Software for 3D Dynamic Analysis Including Soil-Structure Interaction, ACS Version 2.3.0, June 2009, and User Manual ~~SASSI PREP User's Guide,~~ Revision 1~~2~~, August 31, 2009 ~~for ACS SASSI, Version 2.2~~, Ghiocel Predictive Technologies, Inc. Pittsford, NY.

3.7-18 Deleted ~~Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model, MUAP-08005 Rev. 0, April 2008.~~

3.7-19 Design Guidance For Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, Regulatory Guide 1.143, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2001.

3.7-20 Deleted.

3.7-21 ANSYS, Advanced Analysis Techniques Guide, Release 11.0, ANSYS, Inc., 2007

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Table 3.7.1-1 CSDRS Horizontal Acceleration Values and Control Points

Control Point (Hz)	Acceleration (g)
0.5% Damping	
A (50)	0.3
B (12)	1.49
C (2.5)	1.79
D (0.25)	0.22
E (0.1)	0.035
2% Damping	
A (50)	0.3
B (12)	1.06
C (2.5)	1.28
D (0.25)	0.17
E (0.1)	0.028
5% Damping	
A (50)	0.3
B (12)	0.78
C (2.5)	0.94
D (0.25)	0.14
E (0.1)	0.0226
7% Damping	
A (50)	0.3
B (12)	0.68
C (2.5)	0.82
D (0.25)	0.13
E (0.1)	0.021
10% Damping	
A (50)	0.3
B (12)	0.57
C (2.5)	0.68
D (0.25)	0.12
E (0.1)	(0.019)

Notes:

1. 0.3 g PGA
2. Based on RG 1.60, Rev. 1 (Reference 3.7-6) amplification factors
3. For Control Points D & E, acceleration is computed as follows:

Acceleration $= (\varpi^2 D / 386.4 \text{ in/sec}^2) \times F_A \times 0.3$

ϖ = 2π x frequency (rad/sec)

D = Displacement (in)

F_A = Amplification Factor from Regulatory Guide 1.60

Table 3.7.1-2 CSDRS Vertical Acceleration Values and Control Points

Control Point (Hz)	Acceleration (g)
0.5% Damping	
A (50)	0.3
B (12)	1.49
C (3.5)	1.70
D (0.25)	0.15
E (0.1)	0.024
2% Damping	
A (50)	0.3
B (12)	1.06
C (3.5)	1.22
D (0.25)	0.12
E (0.1)	0.018
5% Damping	
A (50)	0.3
B (12)	0.78
C (3.5)	0.89
D (0.25)	0.094
E (0.1)	0.015
7% Damping	
A (50)	0.3
B (12)	0.68
C (3.5)	0.78
D (0.25)	0.086
E (0.1)	0.014
10% Damping	
A (50)	0.3
B (12)	0.57
C (3.5)	0.65
D (0.25)	0.08
E (0.1)	0.012

Notes:

1. 0.3 g PGA
2. Based on RG 1.60, Rev. 1 (Reference 3.7-6) amplification factors
3. For Control Points D & E, acceleration is computed as follows:

$$\text{Acceleration} = (\varpi^2 D / 386.4 \text{ in/sec}^2) \times F_A \times 0.3$$

ϖ = 2π x frequency (rad/sec)

D = Displacement (in)

F_A = Amplification Factor from Regulatory Guide 1.60

Table 3.7.1-4 Comparison of 5% Damping ARS of Synthesized Time History CSDRS

Time History	Frequency Range		0.1 – 1 Hz	1 – 10 Hz	10 – 100 Hz	0.1 – 100 Hz
	No. Freq. Data Points		100	100	100	300
Horizontal H1	ARS/CSDRS ratio	Min.	0.942	0.937	0.920	0.920
		Max	1.251	1.262	1.218	1.262
	Max. No. of Data Point Non-Exceedances Within Any One Particular Frequency Window (1)		1	4	7	7
Horizontal H2	ARS/CSDRS ratio	Min.	0.898	0.94	0.968	0.898
		Max	1.292	1.142	1.126	1.292
	Max. No. of Data Point Non-Exceedances Within Any One Particular Frequency Window (1)		7	6	6	7
Vertical V	ARS/CSDRS ratio	Min.	0.942	0.931	0.966	0.931
		Max	1.206	1.212	1.182	1.212
	Max. No. of Data Point Non-Exceedances Within Any One Particular Frequency Window (1)		6	3	6	6

Note:

1. Maximum number of frequency data points in any one particular sequence (frequency window) for which the acceleration values of the time histories ARS are below those of the CSDRS.

Table 3.7.1-5 Duration of Motion of US-APWR Time Histories with Respect to Arias Intensity

	Arias Intensity		Duration (seconds)
	Time for 5% (seconds)	Time for 75% (seconds)	
H1	3.94	11.46	7.52
H2	4.635	11.78	7.145
V	2.08	10.84	8.77

Table 3.7.2-1 Summary of Dynamic Analysis and Combination Techniques

Summary of Dynamic Analyses & Combination Techniques				
Model	**Analysis Method**	**Program**	**Three Components Combination (for purposes of dynamic analysis)**	**Modal Combination**
Three-dimensional R/B-PCCV-containment internal structure SSI Model [1]	Time History Analysis in Frequency Domain using sub-structuring technique	SASSI	SRSS	N/A
Three-dimensional RCL-R/B-PCCV-containment internal structure FE Model [2]	Time History Analysis in Frequency Domain	ANSYS	N/A [~~1~~2]	N/A
Three-dimensional PS/B SSI Model [3]	Time History Analysis in Frequency Domain using sub-structuring technique	SASSI	SRSS	N/A
Three-dimensional PS/B FE Model [2]	Time History Analysis in Frequency Domain	ANSYS	N/A[2]	N/A

Notes:

1. The three-dimensional RCL-R/B-PCCV-containment internal structure SSI model is addressed in Technical Reports MUAP-10001 and MUAP-10006 (References 3.7-47 and 3.7-48).
2. The FE models for the RCL-R/B-PCCV-containment internal structure on their common basemat and the PS/Bs are used only for validation of the seismic models and for static analysis for design of structural members and components as addressed in Section 3.8.
3. The three-dimensional PS/B model is addressed in Technical Reports MUAP-10001 and MUAP-10006 (References 3.7-47 and 3.7-48).

Table 3.7.2-3 Deleted ~~Seismic SSI Analysis Cases~~

~~Case No.~~	~~Soil Properties~~				~~Input Earthquake (SSE) Time Histories~~		
	~~Soft~~	~~Medium 1~~	~~Medium 2~~	~~Fixed Base~~	~~H1~~	~~H2~~	~~V~~
~~1~~	~~√~~				~~√~~	~~√~~	~~√~~
~~2~~		~~√~~			~~√~~	~~√~~	~~√~~
~~3~~			~~√~~		~~√~~	~~√~~	~~√~~
4				~~√~~	~~√~~	~~√~~	~~√~~

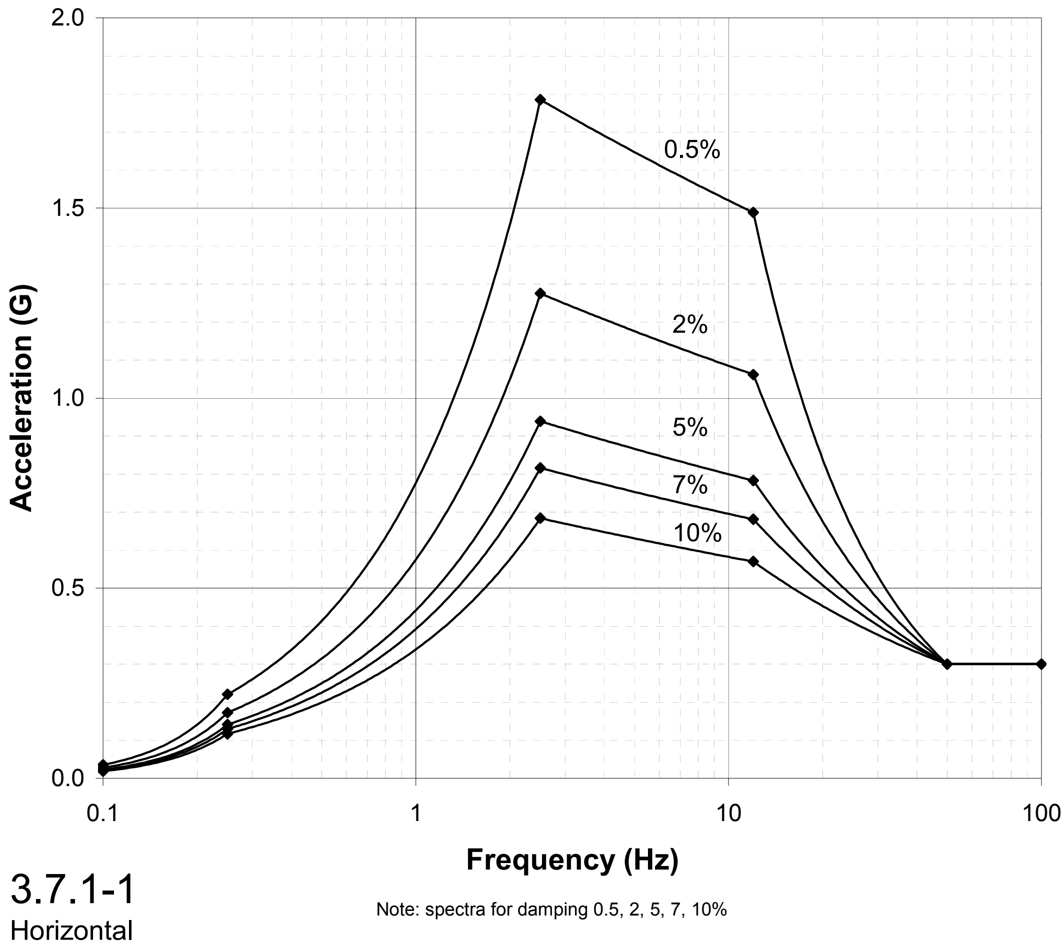

Figure 3.7.1-1 US-APWR Horizontal CSDRS

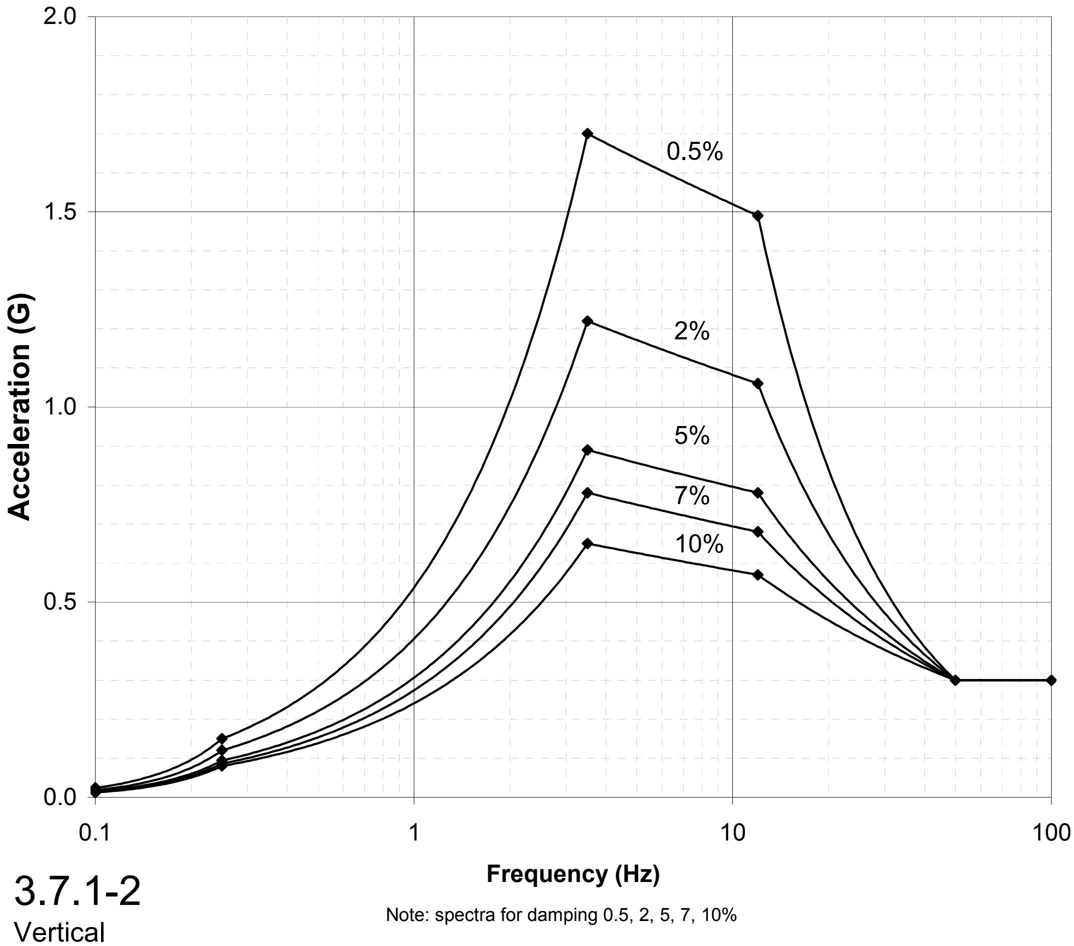

Figure 3.7.1-2 US-APWR Vertical CSDRS

Figure 3.7.2-11 Deleted~~Development of Enveloped Design ISRS~~

Figure 3.7.2-11 Deleted ~~Example Design ISRS~~

- ASME Code, Section III, Subarticle CC-3720 is satisfied by addressing an accident that releases hydrogen generated from 100% fuel clad-coolant reaction accompanied by hydrogen burning, including the effects of temperature and prestress. See Subsection 3.8.1.3.2.2 for further discussion of this design condition.

Load combinations and factors based on ASME Table CC-3230-1 are presented in Table 3.8.1-2. Load combinations involving wind and tornado have been determined to be less severe than other cases through comparison calculations to the design-basis earthquake loads and, therefore, load combinations involving wind and tornado are not used in the full detailed design analyses of the overall PCCV structure and its liner.

3.8.1.3.1 Loads

The following is a brief description of loads unique to the PCCV and liner used in Table 3.8.1-2 for design and analysis. Subsection 3.8.4.3 gives definitions and descriptions of other loads based on the ACI 349 (Reference 3.8-8) and AISC N690-1994, including Supplement 2 (Reference 3.8-9), definitions and descriptions, which are consistent with the ASME Code, Section III.

- **Prestress Load**

  For purposes of the US-APWR PCCV design, prestress is defined as the load on the PCCV dome and cylinder walls that, when applied by mechanical force from tendons after the concrete has hardened, results in the introduction of internal stresses to reduce potential tensile stresses in concrete resulting from other loads. The initial prestress governs the cylinder wall and dome thickness. It is not governed by radiation shielding. The minimum prestress level including all losses after design life applied to the PCCV is 1.20 times the design pressure. The minimum prestress level including all losses after design life applied to the PCCV is 1.20 times the design pressure.

- **Design-Basis Accident Pressure (P_a) and Test Pressure (P_t)**

  The DBA pressure is 68 psig. The DBA pressure is increased for structural design purposes using load factors as shown in ASME Code, Section III, Table CC-3230-1, depending on the particular load combination considered.

  The structural integrity test pressure P_t is 1.15 times the design pressure (P_t = 78.2 psig).

  External or internal events such as containment spray actuation may induce a negative pressure on the PCCV. See Chapter 6 for further discussion. Therefore, the PCCV is designed for a negative pressure of 3.9 psig as a separate event.

  With respect to accident pressure loads, 10 CFR 50.44 (Reference 3.8-10) requires that an analysis be performed that demonstrates that the containment structural integrity is maintained under loads resulting from combustible gases generated from metal-water reaction of the fuel cladding. In determining loads from combustible gases, the US-APWR design follows the guidance of RG 1.7

For the factored load design associated with the prestressed concrete wall:

$$D + P_g1 + [P_g2 \text{ or } P_g3\ ]$$

where

D	=	Dead load
P_g1	=	Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction = 46.7 psia from Reference 3.8-55
P_g2	=	Pressure resulting from uncontrolled hydrogen burning (if applicable) = 127 psia from Reference 3.8-55
P_g3	=	Pressure resulting from post-accident inerting assuming carbon dioxide is the inerting agent (Not applicable to US-APWR)

~~In accordance with RG 1.136 (Reference 3.8-3), a minimum design requirement is maintained by satisfying:~~

~~D + 45 psig~~

~~For the US-APWR, based on a DBA pressure P_a of 68 psig and a corresponding design test pressure of 1.15 x P_a, the above minimum requirement of D + 45 psig is met by virtue of the design and does not require design evaluation.~~

The factored load design of the US-APWR PCCV complies with the guidance of RG 1.136 (Reference 3.8-3). MHI Technical Report MUAP-10018 "US-APWR Containment Performance for Pressure Loads" (Reference 3.8-55) documents the methodology used to determine the pressure effects of an accident that releases hydrogen generated from 100% fuel clad metal-water reaction and uncontrolled hydrogen burning on the PCCV. The maximum pressure considered in the analysis in MUAP-10018 (Reference 3.8-55) is $P_g1 + P_g2$ = 173.7 psia = 159 psig. The analysis also includes effects of dead load D.

3.8.1.3.3 Load Combinations

Load combinations and factors are presented in Table 3.8.1-2, which includes the worst case load combination of dead load, operating live load, and maximum load values of extreme environmental conditions.

3.8.1.3.4 Liner Plate Loads and Load Combinations

Liner plate strains are evaluated for the same loads and load combinations as those used to design the PCCV shell, which are presented in Table 3.8.1-2, except that all load factors for the liner plate are 1.0 in accordance with Subarticle CC-3720 of the ASME Code, Section III (Reference 3.8-2). In general, load cases that are shown to be less severe than other cases do not receive a full design analysis.

Liner plate stresses are evaluated for the construction load category and for the mechanical loads applied to attachments on the liner plate. During construction, the liner plate functions as the inner concrete form and as such it is subject to pressure from concrete placement as a primary load. This pressure can be treated as a hydraulic load

3.8.1.4.2.1 Concrete Cracking Considerations

As discussed in SRP 3.8.1 (Reference 3.8-7) Section II.4.D, concrete cracking can affect the stiffness of the PCCV and cause shifting of the natural frequency, thereby affecting the response/loads used to design the PCCV. Accordingly, the analysis used to calculate the dynamic response of the PCCV resulting from dynamic loads such as earthquake and hydrodynamic loads considers the potential effects of concrete cracking where significant.

The concrete and reinforcement stresses are calculated considering the extent of concrete cracking at these sections. The following are assumptions for calculations:

- The concrete is isotropic and linear elastic but with zero tensile strength
- The thermal forces and moments are reduced according to the concrete cracking depth
- The redistribution of section forces and moments that occurs due to concrete cracking is taken into account

For thermal loads, the effects of concrete cracking are considered in developing the internal forces and moments in the section. For these loads, concrete cracking relieves the thermal stress, as well as redistributes the internal forces and moments on the sections from those obtained from a linear analysis. At the cylinder to basemat junction, cracking reduces the moments since they are created due to self constraint.

3.8.1.4.3 Ultimate Capacity of the PCCV

~~The US-APWR ultimate capacity analysis is based on hand computations using linear methods and comparison to previous test results as explained below. The analysis is considered to be a conservative approach because it does not take credit for redistribution of load or additional strain beyond the yield point of the materials. Although RG 1.136, Rev. 3, Page 10 recommends a non-linear FE analysis to determine the ultimate capacity, the linear analysis approach is considered an acceptable alternative because of its inherent conservatism. SRP 3.8.1, Rev. 2, Page 16, makes a similar recommendation for non-linear FE analysis and suggests estimation of the capacity based on a maximum global hoop strain away from discontinuities of 0.8 %.~~

~~The configuration of the US-APWR PCCV is very similar to the 1/4-scale NUPEC/NRC PCCV Sandia National Laboratories (SNL) model, which was modeled after the latest design of PCCV used in Japan, on which ultimate capacity testing was performed at SNL in New Mexico and documented in Section 3.5 of Reference 3.8-16. That model exhibited functional failure (apparent liner tearing and subsequent leakage) approximately between 2.4 P_d and 2.5 P_d, and ultimate structural failure at approximately 3.6 P_d. Ultimate structural failure was due to membrane rupture of the shell at mid-height of the cylinder (tension hoop strain failure not adjacent to any penetration or discontinuity), with maximum hoop strain of 1.65% at time of ultimate rupture.~~

~~Based on the similarity of the US-APWR PCCV to that of the model tested at SNL, the ultimate structural capacity for the PCCV is estimated conservatively by hand calculations based on the cumulative yield strength of the steel reinforcement, tendons and liner plate acting in membrane hoop tension. The largest material strain capacity of~~

~~the three materials (reinforcement, liner, and tendons) is that of the American Society of Testing and Materials (ASTM) A416 tendons, which have a defined yield strain of 1%. The resulting estimated ultimate pressure capacity of PCCV based on these hand calculations is 201 psig (approximately 3.0 P_d), and is bounded by the SNL model pressure and yield strain test results.~~

The US-APWR ultimate pressure capacity analyses are based on detailed 3D finite element modeling, advanced material constitutive relations including material degradation with temperature, and an assessment of uncertainties within a probabilistic framework.

Accident conditions leading to over-pressurization include elevated temperatures. Because of thermal induced stresses and material property degradation at elevated temperatures, the fragility for over-pressurization is also a function of temperature. Thus, the fragility analyses are conducted for three different thermal conditions, 1) normal operating steady-state conditions, 2) a long term accident condition, and 3) a hydrogen burning condition.

The analyses indicate that the pressure capacity is limited by liner tearing, which is found to first initiate at the transition to the thickened concrete section for the equipment hatch. The expected or median pressure to initiate tearing is found to be 223.6 psig or 3.29 times the design pressure (Pd) of 68 psig for the steady state thermal conditions associated with a long term accident condition. This limitation in pressure capacity due to liner tearing is consistent with the ¼ scale PCCV tests performed at Sandia National Laboratories (SNL), References 3.8-56 and 3.8-57. The 95% confidence value for liner tearing under long term accident conditions is determined to be 176 psig or 2.59* Pd in these analyses. The median capacity due to liner tearing for the hydrogen burning case is found to be 238.5 psig or 3.51* Pd. This pressure is higher than that at normal operating conditions, which is attributed to the compressive stress induced into the liner due to the locally higher temperatures of the liner relative to the concrete.

However, note that the 95% high confidence value for pressure capacity due to liner tearing under hydrogen burning conditions is lower than that for normal operating conditions reflecting the additional uncertainty for the severe accident conditions and effects of high temperatures. For ultimate capacity based on rebar and tendon rupture, the median pressure capacity for long term design accident conditions is found to be 243.6 psig or 3.58* Pd. It is also determined that the ultimate capacity is not limited by the concrete strength. These results are again consistent with the SNL test for the ¼ scale PCCV model. These analyses also indicate that the ultimate capacity does not strongly depend on temperature. The median ultimate capacity at normal operating temperature is determined to be 3.65*Pd and the median ultimate capacity under hydrogen burning conditions is 3.60*Pd.

The fragility analyses and detailed description of the methodologies are summarized in MUAP-10018 (Reference 3.8-55).

Four redundant safety systems containing radioactive material are located in each zone of the four quadrants surrounding the containment structure. Each of the quadrant areas is separated by a physical barrier to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy pipe break events.

Non-radioactive safety systems such as the ESWS, CCWS and electrical system, etc., are located in the plant southern area of the R/B. This area is also separated into four divisions by a physical barrier to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy line break events.

3.8.4.1.2 PS/Bs

The east and west PS/Bs are arranged adjacent to the R/B; one to the east and one to the west. These buildings are free-standing on a reinforced concrete basemat. Each building contains two emergency power sources and one alternate power source which are separated from each other by a physical barrier. In addition, the safety-related chillers are also located in these buildings.

Details of the design and analysis of the east and west PS/Bs are provided in Subsection 3.8.4.4.

3.8.4.1.3 ESWPT, UHSRS, PSFSVs, and Other Site-Specific Structures

The ESWPT is a seismic category I structure constructed of reinforced concrete. Terminating in part under the T/B, the structure is isolated from other structures to prevent any seismic interaction. The other termination point is the UHSRS at the source of the ESWS. [[The UHSRS consist of a [[cooling tower enclosure]], ESWS pump houses, and the UHS [[basin]].]] The PSFSVs are structures which house the safety-related and non safety-related fuel oil tanks.

The design and analysis of the ESWPT, UHSRS, PSFSVs, and other site-specific structures are to be provided by the COL Applicant based on site-specific seismic criteria.

3.8.4.1.4 Heating, Ventilating and Air Conditioning Ducts and Duct Supports

Seismic category I HVAC ducts and duct supports are routed as necessary to supply safety-related functions of air distribution. Appendix 3A describes the qualification of HVAC ducts and duct supports.

3.8.4.1.5 Conduits and Conduit Supports

Seismic category I conduits and conduit supports are routed as necessary to supply safety-related Class-1E cable. The conduit consists of a metal wall of minimum thickness as specific, and is assembled using standard industry fittings and clips. Appendix 3F describes the qualification of conduits and conduit supports.

3.8.4.4.1 R/B

The R/B includes the MCR and the fuel storage area, and is a reinforced concrete structure consisting of vertical shear/bearing walls and horizontal slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs.

The fuel handling area is a reinforced concrete structure supported by structural steel framing. The new fuel is stored in racks in a dry, unlined pit. The spent fuel pit is lined with stainless steel and is normally flooded to an elevation 1 ft, 2 in. below the operating floor deck. Subsection 9.1.2 describes the design bases and layout of the fuel storage area.

The design and analysis procedures for the R/B, other than the PCCV and containment internal structure, including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI-349 (Reference 3.8-8) for concrete structures, with AISC N690 (Reference 3.8-9) for steel structures, and with American Iron and Steel Institute (AISI) specification for cold formed steel structures (Reference 3.8-38).

The design considers normal loads (including construction, dead, live, and thermal), and the SSE. Seismic forces are obtained from the dynamic analysis described in Subsection 3.7.2. These loads are applied to the linear elastic FE model, which extends to the base of the R/B foundation, ~~fixed at elevation 3 ft, 7 in.~~ as equivalent static forces. Soil stiffnesses derived from the standard plant soil profiles are assigned to the subgrade for the design of the overall R/B, and the design of the R/B superstructure is also performed considering a fixed-base condition at the bottom of the foundation. Loads and load combinations are given in Subsection 3.8.4.3.

The design of the R/B’s flexible shear walls and floor slabs, like that of the main steam piping room with many openings, takes into account the out-of-plane bending and shear loads, such as live load, dead load, and seismic load. Also, the walls and slabs of the spent fuel pit and the emergency feedwater pit are designed to resist the out-of-plane bending and shear loads, such as live load, dead load, seismic, hydrostatic, and hydrodynamic pressure.

The R/B is analyzed using a three-dimensional FE model with the ANSYS computer codes (Reference 3.8-14). The FE model is shown in Figure 3.8.4-2.

The basemat design is described in Subsection 3.8.5.

Structural Design of Critical Sections

This subsection summarizes the structural design of representative seismic category I structural elements in the R/B. These structural elements are listed below and the corresponding location numbers are shown on Figure 3.8.4-3.

SECTION 1 West exterior wall of R/B, elevation 3 ft, 7 in. to elevation 101 ft, 0 in. This exterior wall illustrates typical loads such as temperature gradients, seismic, and tornado missile.

SECTION 2 South interior wall of R/B, elevation 3 ft, 7 in. to elevation 101 ft, 0 in. This is one of the most highly stressed shear walls.

3.8.4.7.1 Construction Inspection

Inspections relating to the construction of seismic category I and II SSCs are conducted in accordance with the codes applicable to the construction activities and/or materials. In addition, weld acceptance is performed in accordance with the NCIG, Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01, Revision 2 (Reference 3.8-31).

3.8.5 Foundations

3.8.5.1 Description of the Foundations

Each building is isolated on a separate concrete basemat as identified in Subsection 3.8.4. The PCCV and the containment internal structure are integral with the R/B on a common basemat. Adjoining building basemats, such as the east and west PS/Bs, A/B, and T/B, are structurally separated by a 4 in. gap at and below the grade. This requirement does not apply to engineered mat fill concrete that is designed to be part of the foundation subgrade.

Basemats are located at a depth below the zone of maximum frost penetration, taken as 4 ft below grade. The COL Applicant is to determine if the site-specific zone of maximum frost penetration extends below the depth of the basemats for the standard plant, and to pour fill concrete under any basemat above the frost line so that the bottom of fill concrete is below the maximum frost penetration level.

3.8.5.1.1 Reactor Building and Enveloped Structures

The R/B, with the PCCV and containment internal structure at its center, is built on a common basemat and isolated from the adjacent A/B, east and west PS/Bs, and T/B. The basemat of the R/B is essentially a rectangular-shaped reinforced concrete mat and is composed of two parts. One part of the basemat is for the PCCV and containment internal structure, and the other part is for the remaining seismic category I basemat for the R/B. The length of the basemat in the north-south direction is 309 ft, 0 in., and in the east-west direction is 210 ft, 0 in, as shown in Figure 3J-1. The central region, generally circular with a diameter of approximately 187 ft~~188 ft, 0 in.~~, supports the PCCV and containment internal structure with a thickness of approximately ~~varying from 11 ft, 7 in. to~~ 38 ft, 2 in. The peripheral portion which supports the R/B is 9 ft, 11 in. thick.

The basemat includes hollow portions such as the tendon gallery, tendon gallery access tunnel, and other portions such as in-core chase and CV recirculation sump. Since the vertical tendons are anchored at the roof of the tendon gallery, the upper part of the tendon gallery is important from the structural point of view.

The basemat reinforcement consists of a top horizontal layer of reinforcement, a bottom horizontal layer of reinforcement, and vertical shear reinforcement. The bottom layer of reinforcement is arranged in a rectangular grid. The top layer of reinforcement is arranged in a rectangular grid at the center of the mat and radiates outward in a polar pattern in order to avoid interference with PCCV reinforcement. The top and bottom reinforcement at the upper portion of the tendon gallery is in a polar pattern.

Outlines of the R/B, PCCV and containment internal structure including the basemat are provided in Figures 3.8.5-1 through 3.8.5-3.

3.8.5.1.2 Power Source Buildings

The east and west PS/Bs are free-standing structures, each on an independent reinforced concrete basemat. Each PS/B basemat is a rectangular reinforced concrete mat with a thickness of ~~100~~ 119 in. The bottom of basemat is at elevation -34 ft, 8 in.

The bottom layer of basemat reinforcement is arranged in a rectangular grid. The basemat also consists of a top layer of reinforcement, and vertical shear reinforcement.

3.8.5.1.3 Site Specific Structures

Other non-standard seismic category I plant buildings and structures of the US-APWR are designed by the COL Applicant based on site-specific subgrade conditions.

3.8.5.2 Applicable Codes, Standards and Specifications

The following industry codes, standards and specifications are applicable for the design, construction, materials, testing and inspections of the PCCV basemat. Pressure retention requirements of the vessel are in accordance with the guidance from SRP 3.8.1. (Reference 3.8-7).

- Rules for Construction of Nuclear Facility Components, Division 2, Concrete Containments, Section III, American Society of Mechanical Engineers, 2001 Edition through the 2003 Addenda (hereafter referred to as ASME Code). (Reference 3.8-2).

  Note: Articles CC-1000 through CC-6000 of Section III, Division 2 are acceptable for the scope, material, design, construction, examination, and testing of concrete containments of nuclear power plants subject to the regulatory positions provided by RG 1.136 (Reference 3.8-3).

The following industry standards are applicable for the design and construction of seismic category I basemats not required as a pressure retention boundary. Other codes, standards and specifications applicable to materials, testing and inspections are provided in Subsections 3.8.4.6 and 3.8.4.7.

- ACI 349-01, Code Requirements for Nuclear Safety-Related Concrete Structures, American Concrete Institute, 2001 (Reference 3.8-8)
- RG 1.142, Rev. 2, Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments), U.S. Nuclear Regulatory Commission, Washington, DC, November 2001. (Reference 3.8-19)

3.8.5.3 Loads and Load Combinations

Loads and load combinations are discussed in detail in Subsections 3.8.1.3 and 3.8.4.3. The containment design pressure P_d of 68 psi is included as an accident pressure in these load cases. Other load combinations applicable to the design of the basemat include acceptance criteria for overturning, sliding, and flotation as detailed in Table 3.8.5-1. The non-ASME portion of the basemat is designed in accordance with ACI-349 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19), where applicable. The reinforced concrete basemat for the PCCV and enveloped containment internal structure are designed in accordance with ASME Code Section III, Division 2, Subsection CC (Reference 3.8-2). Figure 3.8.5-4 delineates basemat regions applicable to each Code.

3.8.5.4 Design and Analysis Procedures

Based on the premise that seismic category I buildings basemats are not supported on bedrock, a computer analysis of the SSI is performed for static and dynamic loads. Subsection 3.7.2 provides further information.~~ Two types of SSI analyses are required for the R/B and the PS/Bs: an overall seismic analysis of the building for the superstructure design, and a local analysis of the basemat for its design. For the basemat design, the basemat is modeled using solid finite elements with springs representing the subgrade.~~

The seismic category I structures are concrete, shear-wall structures consisting of vertical shear/bearing walls and horizontal floor slabs designed to SSE accelerations as discussed in Section 3.7. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the loads between them.

The reinforced concrete basemat for the PCCV and enveloped containment internal structure are designed in accordance with ASME Code Section III, Division 2, Subsection CC (Reference 3.8-2). Other seismic category I basemats of reinforced concrete are designed in accordance with ACI-349 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19) where applicable. Table 3.8.5-2 identifies the material properties of concrete and Figure 3.8.5-4 delineates the governing codes based on region of the R/B, PCCV and containment internal structure basemat.

3.8.5.4.1 Properties of Subgrade

For purposes of the US-APWR standard design, the SSI effects are captured by considering three generic subgrade types utilizing frequency independent springs. A fourth subgrade condition is also considered, that of a foundation resting on hard rock. For the fourth condition, it is not necessary to consider SSI effects because the foundation is considered to be resting on a fixed base that is rigid. Subsection 3.7.2.4 provides further discussion relating to SSI and the selection of subgrade types.

The four supporting media (subgrade) conditions for the US-APWR design are provided in Table 3.8.5-3.

The properties of conditions provided in Table 3.8.5-3 are considered to represent stiffness properties of the subgrade material that are compatible to the strains generated in the soil by the input design ground motion. The dissipation of energy in the subgrade media due to the soil material damping is conservatively neglected.

area of the basemat that is uplifted. Minimum area of steel reinforcement is calculated from the section forces for the most critical load combinations.

The required reinforcement steel for the portion of the basemat under the R/B (other than PCCV) is determined by considering the reinforcement envelope for the full non-linear iteration of the most critical load combinations.

3.8.5.4.2.1 Global Three-Dimensional FE Modeling of Basemat

The stress conditions of the basemat for the R/B complex are generated by numerous types of loads from the superstructure. The modeling of the basemat therefore involves evaluating the interaction between the basemat and the superstructures to determine the stress conditions at the interface. The global FE model is analyzed utilizing the FE computer program ANSYS (Reference 3.8-14).

Regarding the R/B, the element divisions in a horizontal direction inside the secondary shield walls of the containment internal structure are ~~is~~ made in a rectangular grid pattern and those divisions ~~that~~ outside the secondary shield wall are ~~is~~ made in a polar pattern. Peripheral areas of the basemat, outside the thickened mat that supports the PCCV and containment internal structure are divided into a rectangular grid.

The upper portion of tendon gallery is considered with concentrated stresses created by the connection with the PCCV. This region is divided into multiple layers of ~~four~~ elements in the radial direction to accommodate the differing concrete strengths in this area as shown schematically in Figure 3.8.5-4 ~~better evaluate the stresses~~.

The basemat below the PCCV and the lower portion of containment internal structure are simulated with ~~hexahedral~~ solid elements (ANSYS SOLID45 elements). The elements below the PCCV are divided into ten layers ~~three to fifteen parts in thickness,~~ and elements in peripheral areas are divided into four layers ~~three parts~~.

The FE modeling of the PS/Bs is addressed ~~provided~~ in Subsection 3.8.4.4.

3.8.5.4.3 Boundary Conditions of Basemat

The basemat subgrade is included in the detailed static FE models used for structural design ~~is represented by translational spring elements that are attached to the bottom of the basemat~~ by meshing a sufficiently large volume of soil/rock below and around the basemat. The stiffness of the backfill around the below-grade walls is not considered in the model. To increase computational efficiency, the subgrade part of the FE model is condensed into a super-element. The properties of the subgrade layers used in the FE model of the subgrade are established based on several profiles selected from the generic layered soil profiles described in Technical Report MUAP-10001 (Reference 3.7-47) to cover the entire range of soil/rock conditions at representative nuclear power plant sites within the central and eastern US. ~~Subgrade coefficients, determined based on the SSI lumped parameter values listed in Table 3H.2-14 of Appendix 3H, are used to assign spring values to the individual nodes of the FE model. These subgrade coefficients are multiplied by the basemat nodal point tributary areas to compute the spring constants assigned to the nodal points. The vertical spring stiffnesses are also developed in a manner such that the cumulative vertical stiffness is equivalent to the vertical SSI spring constant value in Table 3H.2-14.~~

F_h = Lateral force due to active soil pressure, including surcharge, and tornado or wind load, as applicable

The factor of safety against sliding caused by earthquake is identified by the ratio:

FS_{se} = [F_s + F_p] / [F_d + F_h], not less than FS_{sl} as determined from Table 3.8.5-1

where

FS_{se} = Structure factor of safety against sliding caused by earthquake

F_s = Shear (or sliding) resistance along bottom of structure basemat

F_p = Resistance due to maximum passive soil pressure, neglecting any contribution of surcharge. No credit is taken for passive soil pressure in calculating the factor of safety against sliding in standard plant building structures.

F_d = Dynamic lateral force, including dynamic active earth pressures caused by seismic loads

F_h = Lateral force due to all loads except seismic loads

When a coefficient of friction of 0.7 is used in calculating sliding resistance F_S, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required.

3.8.5.5.3 Flotation Acceptance Criteria

The factor of safety against flotation is identified as the ratio of the total dead load of the structure including foundation (D_r) divided by the buoyant force (F_b). Therefore,

FS_f = D_r / F_b , not less than FS_{fl} as determined from Table 3.8.5-1.

where

FS_f = Structure factor of safety against flotation by the maximum design basis flood or ground water table.

D_r = Total dead load of the structure including foundation.

F_b = Buoyant force caused by the design basis flood or high ground water table, whichever is greater.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Subsection 3.8.4.6 describes the materials, quality control, and special construction techniques applicable to seismic category I foundations, including water control structures and below-grade concrete walls and foundations. Subsection 3.8.1.7 provides testing and surveillance requirements relating to the PCCV basemat.

COL 3.8(25) *The site-specific COL are to assure the design criteria listed in Chapter 2, Table 2.0-1, is met or exceeded.*

COL 3.8(26) *Subsidence and differential displacement may therefore be reduced to less than 2 in. if justified by the COL Applicant based on site specific soil properties.*

COL 3.8(27) *The COL Applicant is to specify normal operating thermal loads for site-specific structures, as applicable.*

COL 3.8(28) *The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures.*

COL 3.8(29) *The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.*

COL 3.8(30) *When a coefficient of friction of 0.7 is used in calculating sliding resistance F_s, roughening of fill concrete is required per criteria given in Section 11.7.9 of ACI 349 (Reference 3.8-8). If a coefficient of friction of less than 0.7 is used by the COL Applicant, roughening of fill concrete is not required.*

3.8.7 References

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3.8-18 Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10 Code of Federal Regulations Part 50, Appendix J, U.S. Nuclear Regulatory Commission, Washington, DC.

3.8-19 Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments), Regulatory Guide 1.142, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2001.

3.8-20 Concrete Radiation Shields for Nuclear Power Plants. Regulatory Guide 1.69, U.S. Nuclear Regulatory Commission, Washington, DC, December 1973.

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3.8-51 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, RG 1.143, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2001.

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3.8-53 Cooling and Insulating Systems for Mass Concrete. ACI-207.4R, American Concrete Institute, 2005.

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Table 3.8.4-5 Summary of R/B and PS/Bs Models and Analysis Methods

Computer Program and Model	Analysis Method	Purpose	Concrete Stiffness
~~Three-dimensional ANSYS FE of R/B model fixed at elevation 3 ft, 7 in.~~	~~Static Analysis~~	~~To obtain member forces~~	~~Monolithic~~
Three-dimensional ANSYS FE of R/B whole model	Static Analysis	To obtain member forces ~~for~~ including thermal load	Monolithic(1)
Three-dimensional ANSYS FE of PS/B model	Static Analysis	To obtain member forces	Monolithic(1)

Note:

1. The stress analysis is performed based on the monolithic concrete stiffness, but the thermal stress is reduced by the reduction factor α (=0.5).

EAST-WEST SECTION ELEVATION

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 6 of 7)

3.9.1.1.5.1 Primary-Side Hydrostatic Test

Both factory and plant site hydrostatic tests occur as a result of component or system testing. This hydrostatic test is performed at a water temperature compatible with reactor material ductility requirements and a test pressure of 3,107 psig (1.25 times design pressure). This transient is assumed to occur five times during the plant design life.

3.9.1.1.5.2 Secondary-Side Hydrostatic Test

The secondary side of the SG is pressurized to 1.25 times the design pressure, with a minimum water temperature of 120°F. This transient is assumed to occur five times during the plant design life.

3.9.1.2 Computer Programs Used in Analyses

3.9.1.2.1 List of Programs

A number of computer programs are used for static, dynamic, and hydraulic transient analysis. The computer programs used in piping analysis are listed in Section 3.12. The following is a list of programs used for the mechanical system component analysis.

- Abaqus — FE structural analysis program (Reference 3.9-6)
- ANSYS — FE structural analysis program (Reference 3.9-7)
- RELAP-5 — Transient hydraulic analysis program (Reference 3.9-8)
- MULTIFLEX — Thermal-hydraulic-structural system analysis program (Reference 3.9-9)
- NASTRAN — FE structural analysis program (Reference 3.9-10)
- ~~SQUIRT — Analysis for leakage rate and area of crack opening for cracked pipe (Reference 3.9-11)~~

3.9.1.2.2 Program Validations

The verification and validation of computer programs is performed in compliance with the established quality assurance program (QAP). Error reporting and resolution of the errors are tracked following a QAP. The QAP is described in Chapter 17. The computer programs are validated using one of the methods described below. Verification tests demonstrate the capability of a computer program to produce valid results for the test problems encompassing the range of permitted usages defined by the program documentation.

- Hand calculations
- Known solution for similar or standard problem
- Acceptable experimental test results
- Published analytical results

3.9.2.1.1.2 ASME Class 2 and 3 or ANSI B31.1 Piping

For ASME Class 2 and 3 or ANSI B31.1 piping (Reference 3.9-14):

$$S_{alt} = \frac{C_2 K_2}{Z} M \leqslant \frac{S_{el}}{\propto}$$

where

C_2K_2 = $2i$

i = stress intensification factor, as defined in Subsection NC-3600 and ND-3600 of the ASME Code (Reference 3.9-1) or in ANSI B31.1 (Reference 3.9-14).

The results of the vibration-induced stresses will determine if support or system modifications are necessary. Based on the level of vibration, the design specification may be modified using the measured vibration as input to assure conformance to applicable codes. If modifications are performed, then the system is retested until acceptable results are achieved.

3.9.2.1.2 System Thermal Expansion Program

The system thermal expansion testing program verifies that the piping expands within acceptable limits during heatup and cooldown. In addition, a review of manufactured standard supports, such as spring hangers, snubbers and struts, is completed to verify thermal expansion is accommodated within acceptable limits during various operational modes. System thermal expansion tests are developed in accordance with the guidance of ASME OM (Reference 3.9-13), Part 7. If piping system restraints are determined during the test to be inadequate or are damaged, corrective restraints are to be installed and another test performed to determine whether the thermal motion has been reduced to an acceptable level. The detailed description of the thermal motion monitoring program will be included as part of the ITP plan. The thermal motion monitoring program will include verification of snubber movement, adequate clearances and gaps, the acceptance criteria, and how the motion is to be measured.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

3.9.2.2.1 Seismic Qualification Testing

Seismic category I mechanical equipment and supports are designed to safely withstand the effects of postulated earthquakes combined with appropriate effects of normal and accident conditions without loss of intended safety-related function. Seismic qualification is performed by either analysis, testing or by a combination of both testing and analysis. The methods for seismic qualification of safety-related mechanical equipment by testing is performed in accordance with the recommendations of “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations”, ANSI/IEEE Std 344-2004~~1987~~ (Reference 3.9-34~~15~~), as endorsed by NRC, RG 1.100, Rev. 3~~2~~, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (Reference 3.9-16). The seismic qualification testing methods for safety-related mechanical equipment are described in Subsection 3.10.2.

3.9.2.2.2 Seismic System Analysis Methods

The seismic system and subsystem analysis methods (including response spectrum analysis, time history analysis, and equivalent static load analysis) are discussed in Subsections 3.7.2 and 3.7.3. The method of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Std 344-2004~~1987~~ (Reference 3.9-34~~15~~) and Subsections 3.10.2 and 3.10.3. The majority of mechanical equipment is supplied by vendors that are required to provide a seismic qualification report that meets the design specification provided in the purchase order.

The stiffness of the seismic subsystem anchorage must be determined and the assumptions made in the seismic analysis must be verified as accurately reflecting the mounting condition.

Two separate models are used for the RCL seismic analysis. One for RCL seismic analysis, which consists of the use of stick mass spring model of SG, RCP, Reactor Pressure Vessel, loop piping and buildings. The other is used for seismic analysis of internal components of the SG. RCL seismic analysis is described in Appendix 3C. The SG seismic analysis is performed considering internal components.

3.9.2.2.3 Determination of Number of Earthquake Cycles

The OBE is chosen as 1/3 of the SSE for the US-APWR (see Subsection 3.7.1.1). When the OBE is defined as less than or equal to 1/3 SSE, explicit design or analysis is not required for the OBE.

With the elimination of OBE, to account for fatigue in analysis and testing, the guidance for determination of the number of earthquake cycles described in “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”, NRC Staff Requirements Memorandum (SRM) SECY-93-087 (Reference 3.9-17) is used. The number of earthquake cycles is discussed in Subsection 3.7.1.1 and Subsection 3.10.2. For piping analysis, the guidance in SECY-93-087 is used and the number of earthquake cycles to consider is defined in Section 3.12, Table 3.12-2, Note 3.

3.9.2.2.4 Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should be preferably less than one half or more than twice the dominant frequencies of the forcing frequencies of the support structure. When the equipment frequencies are within this range, the equipment must be designed for the applicable loads.

3.9.2.2.5 Three Components of Earthquake Motion

The combination of three components of earthquake motion is dependent on the method used in the seismic analysis and is in accordance with “Combining Modal Responses and Spatial Components in Seismic Response Analysis”, RG 1.92, Rev.2 (Reference 3.9-18) as discussed in Subsection 3.7.3.4.

location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment and allowing for unrestricted thermal growth.

Snubbers are modeled as stiffness elements in the piping system seismic stress analysis. Under thermal growth conditions, there is little stiffness; however, under sudden dynamic motion the stiffness is large.

With the implementation of LBB criteria and the elimination of the analysis of dynamic effects of pipe breaks detailed in Subsection 3.6.3, the use of snubbers is minimized in LBB qualified piping systems.

Snubber operability determination is considered part of the ASME Code, Section XI (Reference 3.9-43) inspection program. The ASME OM Code (Reference 3.9-13) is used to develop the inservice testing plan.

3.9.4 Control Rod Drive Systems

3.9.4.1 Descriptive Information of CRDS

The control rod drive system (CRDS) provides one of the independent reactivity control systems, driving a rod cluster control ~~cluster~~ assembly (RCCA) which consists of 24 rodlets and acts as a neutron absorber in the reactor core.

The CRDM inserts and withdraws the RCCA, thus, adjusting the core output. It is operated and controlled by the CRDM control system.

3.9.4.1.1 CRDM

The CRDM for the US-APWR is of the magnetically operated jacking type, which is based on the L-106A type CRDM which has been used in many operating plants in the United States and Japan. These design improvements do not affect operability.

The CRDM consists of a pressure housing, latch assembly, drive rod assembly, and coil stack assembly. The pressure housing involves the latch assembly and the drive rod assembly, and supports the coil stack assembly on the outside of the pressure housing.

The coil stacks are energized sequentially by electric power, which cause magnetic power in the latch assembly to move the latches. The latches hold and move the drive rod assembly which is connected with the RCCA. When de-energized, the coil stacks and the latches release the drive rod assembly. Then the RCCA is inserted within the core by gravity.

The position of the RCCA is measured by the rod position indicator assembly, which is mounted surrounding the rod travel housing. The rod position indicator assembly includes discrete coils which magnetically sense the ferromagnetic drive rod assembly when it moves through the coil centerline. This system is further described in Subsection 7.7.1.4.

A total of 69 CRDMs are mounted on top of the RV closure head and installed directly to the closure head in a vertical position.

readiness as set forth in 10 CFR 50.55a(f) (Reference 3.9-29) and ASME OM Code (Reference 3.9-13).

The pumps covered in the IST Program are those pumps that are provided with an emergency power source and required to perform a specific function in shutting down a reactor to a safe-shutdown condition, in maintaining the safe-shutdown condition, or in mitigating the consequence of an accident.

The US-APWR utilizes ASME OM Code (Reference 3.9-13) for developing the IST Program for ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints. The COL Applicant is to administratively control the edition and addenda to be used for the IST program plan, and to provide a full description of their IST program plan for pumps, valves, and dynamic restraints.

3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

IST of ASME Code, Section III, Class 1, 2, and 3 pumps, valves and dynamic restraints is performed in accordance with the ASME OM Code and applicable addenda, as required by 10 CFR 50.55a(f), except where specific relief has been granted by the NRC in accordance with 10 CFR 50.55a(f). The IST program assesses and verifies operational readiness included in various sections of the ASME OM Code as follow:

- Requirements for IST of pumps are incorporated in ISTB.
- Requirements for IST of valves are incorporated in ISTC.
- Requirements for IST of pressure relief valves are incorporated in Appendix I.
- Requirements for IST of dynamic restraints are incorporated in ISTD.

The various provisions for testing pumps, valves, and dynamic restraints are incorporated into the design of the US-APWR. These provisions and requirements are discussed in Section 3.10 of the DCD.

It should be noted that the requirements of system pressure test per ASME Code, Section XI, Section IWA 5000 (Reference 3.9-43) that verify the system pressure boundary integrity are part of the ISI Program and are not part of this IST Program.

As required by the 10 CFR 50.55a(f) (Reference 3.9-29), ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints are incorporated in 120-month interval IST Program Plan that is in compliance with the requirements of the latest edition and addenda of the OM Code, 12 months before initial fuel load ~~the date of issuance of the operating license~~ and, in compliance with plant~~Plant,~~ Technical Specifications and this DCD. The requirements for the IST Program are included in Technical Specification Subsection 5.5.8 of Section 5.5, Programs and Manuals.

The IST Program Plan is also used for the required preservice (base line) testing of ASME Code, Section III, Class 1, 2, and 3 safety-related pumps, valves and dynamic restraints.

NCA, NB, NC, ND, NF, NG, Code Cases and Appendices including Appendix I, F, and N, 2001 edition thru 2003 Addenda.[4]

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Table 3.9-3 ~~Minimum~~ Design Loading Combinations for ASME Code, Section III, Class 1, 2, 3 and CS Systems and Components

ASME Service Level	Design Loading Combinations(3)(6)
Design	$P + DL + L_{DM} + L_{EM}$
Level A	$P_M{}^{(1)} + DL + L_{EM}$
	$P_M{}^{(1)} + DL + L_{DFN}{}^{(7)} + L_{EM}{}^{(7)} + TH_{TRN} + TH_{MTL}$
Level B	$P_M{}^{(1)} + DL + L_{EM}{}^{(7)} + TH_{TRN} + TH_{MTL} + SRSS^{(2)}((SSEI + SSEA)^{(11)} + L_{DFU}{}^{(7)})$
Level C	$P_M{}^{(1)} + DL + L_{DFE}{}^{(7)} + L_{EM}{}^{(7)}$
	$P_M{}^{(1)} + DL + L_{DF} + L_{EM}{}^{(8)}$
Level D	$P_M{}^{(1)} + DL + L_{DFF}{}^{(7)} + L_{EM}{}^{(7)}$
	$P_M{}^{(1)} + DL + SRSS^{(2)}((SSEI + SSEA) + DBPB) + L_{EM}{}^{(4)}$
	$P_M{}^{(1)} + DL + RV_{OS} + SRSS^{(2)}(SSEI + SSEA) + L_{EM}{}^{(9)}$
	$P_M{}^{(1)} + DL + L_{DFS} + SRSS^{(2)}((SSEI + SSEA) + DBPB + L_{DF}) + L_{EM}{}^{(8)}$
Hydrostatic Test	$H_{DL}{}^{(10)}$

Notes:

1. P_M is the maximum operational pressure for various ASME service levels of operation and dependent on the type of transient that occurs at a particular service level. During an earthquake P_M is considered normal operational pressure at 100% power levels.
2. SRSS sums the squares of each load and determines the resultant square root.
3. Loadings generated by static displacement of the concrete containment vessel and building settlement are added to the loading combinations for ASME Code, Section III, Class 2 and 3 systems.
4. When determining appropriate load combinations involving L_{EM}, a determination of the timing sequence and initiating conditions that occur between P_M and L_{EM} are considered.
5. Deleted.
6. Table 3.9-5 provides a description of loads listed in this table.
7. In determining service level A, B, C, and D load combinations, the timing sequence and initiating conditions that occur between P_M, L_{DFN}, L_{DFU}, L_{DFE}, L_{DFF}, and L_{EM}, are considered respectively.
8. In determining appropriate service level load combination, the timing sequence and initiating conditions that occur between P_M, L_{DF}, and L_{EM}, are considered.
9. In determining appropriate service level load combination, the timing sequence and initiating conditions that occur between P_M, RV_{OS}, and L_{EM}, are considered.
10. If, during operation, the system normally carries a medium other than water (air, gas, steam), sustained loads should be checked for weight loads during hydrostatic testing as well as normal operation weight loads.
11. The earthquake inertial and anchor movement loads used in the Level B Stress Intensity Range and Alternating Stress calculations are taken as 1/3 of the peak SSE inertial and anchor movement loads or as the peak SSE inertial and anchor movement loads. If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading are 300 as derived in accordance with ~~Appendix D of~~ Institute of Electrical and Electronic Engineers Standard 344-2004~~1987~~ (Reference 3.9-34~~15~~). If the earthquake loads are taken as the peak SSE loads then 20 cycles of earthquake loading are considered.

Table 3.9-4 ~~Minimum~~ Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components

Condition	Design Loading Combinations[3]
Design	$DL + L_{DM}$
Level A Service	$DL + TH_i + L_{EM} + L_{DFN}^{(4)} + F$
Level B Service	$DL + TH_i + L_{EM} + L_{DFU}^{(4)}$
Level C Service	$DL + TH_{E\cancel{i}} + L_{EM} + L_{DFE}^{(4)}$
Level D Service	$DL + TH_i + L_{EM} + RV_{OS} + SSEI + SSEA + SE^{(6)(8)}$
	$DL + TH_{F\cancel{i}} + L_{EM} + L_{DFF}^{(4)}$
	$DL + TH_i + L_{EM} + SRSS\ (DBPB + (SSEI + SSEA + SE))^{(6)}$
	$DL + TH_i + L_{EM}^{(7)} + L_{DFS} + SRSS\ (DBPB + (SSEI + SSEA + SE))^{(6)} + L_{DF}^{(7)}$
Hydrostatic Test	$H_{DL}^{(9)}$

Notes:

1. SRSS sums the squares of each load and determines the resultant square root.
2. Deleted.
3. Table 3.9-5 provides a description of loads listed in this table.
4. In determining service level A, B, C, and D load combinations, the timing sequence and initiating conditions that occur between TH_i, L_{DFN}, L_{DFU}, L_{DFE}, L_{DFF}, and L_{EM}, are considered respectively.
5. Deleted.
6. *SE* is support self weight excitation of the support, caused by seismic building inertial loads. *SSEI*, *SSEA*, and *SE* are combined using absolute summation.
7. In determining appropriate service level load combination, the timing sequence and initiating conditions that occur among TH_i and L_{DF} are considered.
8. In determining appropriate service level load combination, the timing sequence and initiating conditions that occur among TH_i and RV_{OS} are considered.
9. If, during operation, the system normally carries a medium other than water (air, gas, steam), sustained loads should be checked for weight loads during hydrostatic testing as well as normal operation weight loads.

Table 3.9-5 ASME Code, Section III, Class 1, 2, 3, CS, and Support Load Symbols and Definitions (Sheet 1 of 2)

Load Symbol	Load Definition
DL	Dead Load (The dead weight consists of the weight of the piping, structures, components, insulation, and other loads permanently imposed upon the piping)
P	Design Pressure, psi
P_M	Maximum service pressure, psi
F	Friction Loads
TH_i	Thermal Loading for ASME Service Conditions, i = applicable service level (TH_N, TH_U, TH_{MTL}, see below) TH_N = ASME Service Level A (Normal) Thermal Loads TH_U = ASME Service Level B (Upset) Thermal Loads TH_{MTL} = ASME Service Level A (Normal) and Level B(Upset) Miscellaneous Thermal Loads with Thermal Stratification and Thermal Cycling Effects
TH_{TRN}	Thermal Transient Load
TH_E	ASME Service Level C (Emergency) Thermal Load
TH_F	ASME Service Level D (Faulted) Thermal Load
L_{EM}	External Mechanically Applied Loads, Including Equipment Nozzle-to-Pipe Reactions
L_{DM}	Design Mechanical Loads other than *DL*. This includes Service Level A Loads and Open Relief Valve Dynamic Loads that are Service Level B
SSEI	Safe-shutdown Earthquake Inertia Loads
SSEA	Safe-shutdown Earthquake Anchor Loads Other Than Pipe Reactions
L_{DF}	Dynamic Loads (Transient Valve Loads including Qvc (Quick Valve Closure), RV_C (Relief Valve Closed System Sudden Opening), RV_O (Relief Valve Open System Sudden Opening) associated with ASME Level A, B, C, and D Service Conditions
L_{DFS}	Sustained Dynamic Loads Associated with ASME Level A, B, C, and D Service Conditions
L_{DFN}	ASME Service Level A (Normal) Dynamic Loads (Transient Valve Loads including QV_C (Quick Valve Closure), RV_C (Relief Valve Closed System Sudden Opening), RV_O (Relief Valve Open System Sudden Opening)
L_{DFU}	ASME Service Level B (Upset) Dynamic Loads (Transient Valve Loads including QV_C (Quick Valve Closure))
L_{DFE}	ASME Service Level C (Emergency) Dynamic Loads (Transient Valve Loads including QV_C (Quick Valve Closure))
L_{DFF}	ASME Service Level D (Faulted) Dynamic Loads (Transient Valve Loads including QV_C (Quick Valve Closure))
SE	*SE* is Support self weight excitation, the effect of the acceleration of the support mass caused by building inertial loads such as *SSEI*

Table 3.9-11 Core Support Structures and Threaded Structural Fasteners Loading Conditions and Load Combinations

	Service Limits for Load Combinations				
Loading Conditions	**Design**	**Level A(5)**	**Level B**	**Level C(1)**	**Level D(1)**
Pressure differences	X(2)	X	X	n/a(3)	n/a
Weights and buoyant forces	X	X	X	X	X
Lift and drag flow loads	X	X	X	X	X
Reactor internals hold down spring load	X	X	X	X	X
Superimposed internal loads from fuel assembly (f/a) hold-down spring forces; f/a lift forces; f/a weights and buoyant forces; f/a grid loads from SSE and LOCA	X	X	X	X	X
1/3 SSE loads(4)	n/a	n/a	X	n/a	n/a
SSE loads	n/a	n/a	n/a	n/a	X
External reaction loads from restraints at upper core support flange; core barrel flange; core barrel nozzle; radial support keys	n/a	X	X	X	X
Thermal loads from transients, gamma heating, and differential thermal expansion	n/a	X	X	n/a	n/a
LOCA differential pressure loads	n/a	n/a	n/a	X	X
Vibratory loads from flow induced vibration and pump induced vibration	X	X	X	n/a	n/a
RCCA stuck rod load (analyzed separately)	n/a	X	n/a	n/a	n/a
Handling and shipping loads (analyzed separately)	n/a	X	n/a	n/a	n/a

Notes:

1. Level C loads and Level D loads are combined for SSE and LOCA (LBB) by SRSS.
2. The letter X means stresses to be combined and compared to the design or service stress limits. For example, Level B service limits in the ASME Code, Sub-section NG should be larger than the core support structures stress intensity and fatigue usage factors resulting from the combined loadings of pressure difference + weight and buoyant forces + lift and drag forces + external reaction restraints + thermal loads + vibratory loads.
3. n/a means not applicable or not required to be analyzed.
4. The earthquake inertial loads used in the Level B Stress Intensity Range and Alternating Stress calculations are taken as 1/3 of the peak SSE inertial loads. The number of cycles to be considered for earthquake loading are 300 as derived in accordance with ~~Appendix D of~~ Institute of Electrical and Electronic Engineers Standard 344-2004~~1987~~ (Reference 3.9-34~~16~~).
5. The Hot Functional Test Condition can be enveloped by the limits for Level A.

Table 3.9-14 Valve Inservice Test Requirements (Sheet 7 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RCS-VLV-140	Vacuum venting line check valve bypass	Manual	Maintain Close	Passive Containment Isolation Safety Seat Leakage	A	Containment Isolation Leak Test	5
~~RCS-VLV-167~~	~~Nitrogen gas supply line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
	Pressurizer relief tank rupture disk	Rupture Disk	Transfer Open	Active	D	Device replacement/ 5 Years	
	Pressurizer relief tank rupture disk	Rupture Disk	Transfer Open	Active	D	Device replacement/ 5 Years	
CVS-AOV-001A	Letdown valve	Remote AO Globe	Transfer Close Maintain Close	Active-to-Fail Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Operability Test	4

Table 3.9-14 Valve Inservice Test Requirements (Sheet 15 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CVS-AOV-192D	Reactor coolant pump seal return line isolation	Remote AO Globe	Maintain Close Transfer Close	Active-to-Fail Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-AOV-196A	Reactor coolant pump seal return line isolation	Remote AO Globe	Maintain Close Transfer Close	Active-to Fail Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-AOV-196B	Reactor coolant pump seal return line isolation	Remote AO Globe	Maintain Close Transfer Close	Active-to Fail Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-AOV-196C	Reactor coolant pump seal return line isolation	Remote AO Globe	Maintain Close Transfer Close	Active-to Fail Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7

Table 3.9-14 Valve Inservice Test Requirements (Sheet 17 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CVS-LCV~~121B~~-031B	Volume control tank outlet valve	Remote MO Gate	Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-LCV~~121C~~-031C	Volume control tank outlet valve	Remote MO Gate	Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-LCV~~121D~~-031D	Charging pump alternate makeup valve	Remote MO Gate	Transfer Open Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-LCV~~121E~~-031E	Charging pump alternate makeup valve	Remote MO Gate	Transfer Open Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7

Table 3.9-14 Valve Inservice Test Requirements (Sheet 18 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CVS-LCV~~121F~~-031F	Charging pump alternate makeup valve	Remote MO Gate	Transfer Open Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-LCV~~121G~~-031G	Charging pump alternate makeup valve	Remote MO Gate	Transfer Open Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/ Cold Shutdown Operability Test	7
CVS-VLV-125	Volume control tank outlet check	Check	Maintain Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-129A	Charging pump minimum flow check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-129B	Charging pump minimum flow check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-595	Charging pump alternate makeup line check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements (Sheet 19 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CVS-VLV-592	Charging pump alternate makeup line check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-594	Charging pump alternate makeup line check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-131A	Charging pump discharge check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-131B	Charging pump discharge check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-181A	Reactor coolant pump seal injection line check (First)	Check	Maintain Open Transfer Open Transfer Close Maintain Close	Active RCS Pressure Boundary	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-181B	Reactor coolant pump seal injection line check (First)	Check	Maintain Open Transfer Open Transfer Close Maintain Close	Active RCS Pressure Boundary	BC	Check Exercise/ Refueling Outage	3
CVS-VLV-181C	Reactor coolant pump seal injection line check (First)	Check	Maintain Open Transfer Open Transfer Close Maintain Close	Active RCS Pressure Boundary	BC	Check Exercise/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 22 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CVS-VLV-202	Reactor coolant pump seal return line containment isolation check	Check	Maintain Close	Passive Containment Isolation Safety Seat Leakage	AC	Containment isolation Leak Test	5
~~CVS-VLV-653~~	~~Charging line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment isolation Leak Test~~	~~5~~
~~CVS-VLV-667A~~	~~Reactor coolant pump seal injection line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment isolation Leak Test~~	~~5~~
~~CVS-VLV-667B~~	~~Reactor coolant pump seal injection line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 23 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~CVS-VLV-667C~~	~~Reactor coolant pump seal injection line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment isolation Leak Test~~	~~5~~
~~CVS-VLV-667D~~	~~Reactor coolant pump seal injection line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment isolation Leak Test~~	~~5~~
SIS-MOV-001A	Safety injection pump suction isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage	
SIS-MOV-001B	Safety injection pump suction isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage	
SIS-MOV-001C	Safety injection pump suction isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage	

Table 3.9-14 Valve Inservice Test Requirements (Sheet 24 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
SIS-MOV-001D	Safety injection pump suction isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage	
SIS-VLV-004A	Safety injection pump discharge check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
SIS-VLV-004B	Safety injection pump discharge check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
SIS-VLV-004C	Safety injection pump discharge check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
SIS-VLV-004D	Safety injection pump discharge check	Check	Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
SIS-MOV-009A	Safety injection pump discharge containment isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage Exercise Full Stroke/ Quarterly Operability Test	
SIS-MOV-009B	Safety injection pump discharge containment isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage Exercise Full Stroke/ Quarterly Operability Test	

Table 3.9-14 Valve Inservice Test Requirements (Sheet 25 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
SIS-MOV-009C	Safety injection pump discharge containment isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage Exercise Full Stroke/ Quarterly Operability Test	
SIS-MOV-009D	Safety injection pump discharge containment isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Remote Position	A	Leak Test/ Refueling Outage Exercise Full Stroke/ Quarterly Operability Test	
SIS-VLV-010A	Safety injection pump discharge containment isolation check	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	AC	Leak Test/ Refueling Outage Check Exercise / Refueling Outage	3
SIS-VLV-010B	Safety injection pump discharge containment isolation check	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	AC	Leak Test/ Refueling Outage Check Exercise / Refueling Outage	3
SIS-VLV-010C	Safety injection pump discharge containment isolation check	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	AC	Leak Test/ Refueling Outage Check Exercise / Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 26 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
SIS-VLV-010D	Safety injection pump discharge containment isolation check	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	AC	Leak Test/ Refueling Outage Check Exercise / Refueling Outage	3
SIS-MOV-011A	Direct vessel safety injection line isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close Transfer Open	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
SIS-MOV-011B	Direct vessel safety injection line isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close Transfer Open	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
SIS-MOV-011C	Direct vessel safety injection line isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close Transfer Open	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	

Table 3.9-14 Valve Inservice Test Requirements (Sheet 27 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
SIS-MOV-011D	Direct vessel safety injection line isolation	Remote MO Globe	Maintain Open Maintain Close Transfer Close Transfer Open	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
SIS-VLV-012A	Direct vessel injection line check	Check	Maintain Close Transfer Open	Active RCS Pressure Boundary Safety Seat Leakage	AC	Check Exercise/Refueling Outage Pressure Isolation Leak Test/ Refueling Outage	3
SIS-VLV-013A	Direct vessel injection line check	Check	Maintain Close Transfer Open	Active RCS Pressure Boundary Safety Seat Leakage	AC	Check Exercise/Refueling Outage Pressure Isolation Leak Test/ Refueling Outage	3
SIS-VLV-012B	Direct vessel injection line check	Check	Maintain Close Transfer Open	Active RCS Pressure Boundary Safety Seat Leakage	AC	Check Exercise/Refueling Outage Pressure Isolation Leak Test/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements (Sheet 38 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
SIS-SRV-126C	Accumulator safety valve	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
SIS-SRV-126D	Accumulator safety valve	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
~~SIS-VLV-058A~~	~~Safety injection pump discharge containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~SIS-VLV-058B~~	~~Safety injection pump discharge containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~SIS-VLV-058C~~	~~Safety injection pump discharge containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~SIS-VLV-058D~~	~~Safety injection pump discharge containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 39 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~SIS-VLV-156~~	~~Accumulator nitrogen supply containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
RHS-MOV-001A	Containment spray/residual heat removal pump hot leg isolation – Inner	Remote MO Gate	Maintain Close Transfer Close Transfer Open	Active RCS Pressure Boundary Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/ Refueling Outage Operability Test	8
RHS-MOV-002A	Containment spray/residual heat removal pump hot leg isolation – Outer	Remote MO Gate	Maintain Close Transfer Close Transfer Open	Active RCS Pressure Boundary Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Pressure Isolation Leak Test/ Refueling Outage Operability Test	8 10

Table 3.9-14 Valve Inservice Test Requirements (Sheet 46 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RHS-SRV-023B	Containment spray/residual heat removal heat exchanger outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
RHS-SRV-023C	Containment spray/residual heat removal heat exchanger outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
RHS-SRV-023D	Containment spray/residual heat removal heat exchanger outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
RHS-MOV-025A	Containment spray/residual heat removal pump full-flow test line stop	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	

Table 3.9-14 Valve Inservice Test Requirements (Sheet 47 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RHS-MOV-025B	Containment spray/residual heat removal pump full-flow test line stop	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
RHS-MOV-025C	Containment spray/residual heat removal pump full-flow test line stop	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
RHS-MOV-025D	Containment spray/residual heat removal pump full-flow test line stop	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
RHS-MOV-026A	Residual heat removal flow control	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Operability Test	8

Table 3.9-14 Valve Inservice Test Requirements (Sheet 48 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RHS-MOV-026B	Residual heat removal flow control	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Operability Test	8
RHS-MOV-026C	Residual heat removal flow control	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Operability Test	8
RHS-MOV-026D	Residual heat removal flow control	Remote MO Globe	Maintain Close Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Cold Shutdown Operability Test	8
RHS-VLV-027A	Residual heat removal(RHR) discharge line check	Check	Maintain Close Transfer Open	Active RCS Pressure Boundary Safety Seat Leakage	AC	Check Exercise/Refueling Outage Pressure Isolation Leak Test/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements (Sheet 51 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RHS-VLV-004B	Containment spray/residual heat removal pump suction line check	Check	Maintain Close Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
RHS-VLV-004C	Containment spray/residual heat removal pump suction line check	Check	Maintain Close Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
RHS-VLV-004D	Containment spray/residual heat removal pump suction line check	Check	Maintain Close Transfer Open	Active	BC	Check Exercise/ Refueling Outage	3
~~RHS-VLV-062A~~	~~Containment spray/residual heat removal pump discharge line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 52 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~RHS-VLV-062B~~	~~Containment spray/residual heat removal pump discharge line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~RHS-VLV-062C~~	~~Containment spray/residual heat removal pump discharge line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~RHS-VLV-062D~~	~~Containment spray/residual heat removal pump discharge line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

**Table 3.9-14 Valve Inservice Test Requirements
(Sheet 55 of 151)**

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
EFS-MOV-103A	Turbine driven emergency feed water pump steam inlet	Remote MO Gate	~~Maintain Close~~ Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
EFS-MOV-103D	Turbine driven emergency feed water pump steam inlet	Remote MO Gate	Transfer Open Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
EFS-MOV-101A	Turbine driven emergency feed water pump steam supply line isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
EFS-MOV-101B	Turbine driven emergency feed water pump steam supply line isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 56 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
EFS-MOV-101C	Turbine driven emergency feed water pump steam supply line isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
EFS-MOV-101D	Turbine driven emergency feed water pump steam supply line isolation	Remote MO Gate	Maintain Open Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	B	Remote Position Indication, Exercise/ 2 Years Exercise Full Stroke/Quarterly Operability Test	
EFS-VLV-008A	Emergency feed water pit outlet check	Check	Transfer Open ~~Transfer Close~~	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-008B	Emergency feed water pit outlet check	Check	Transfer Open ~~Transfer Close~~	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-012A	Emergency feed water pump discharge check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-012B	Emergency feed water pump discharge check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements (Sheet 58 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
EFS-VLV-018A	Emergency feed water feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	C	Check Exercise/ Refueling Outage	3
EFS-VLV-018B	Emergency feed water feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	C	Check Exercise/ Refueling Outage	3
EFS-VLV-018C	Emergency feed water feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	C	Check Exercise/ Refueling Outage	3
EFS-VLV-018D	Emergency feed water feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	C	Check Exercise/ Refueling Outage	3
EFS-VLV-102A	Emergency feed water pump steam feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-102B	Emergency feed water pump steam feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-102C	Emergency feed water pump steam feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3

Table 3.9-14 Valve Inservice Test Requirements (Sheet 59 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
EFS-VLV-102D	Emergency feed water pump steam feeding line check	Check	Transfer Open Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
EFS-VLV-109A	Turbine driven emergency feedwater pump steam supply drain line check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	
EFS-VLV-109B	Turbine driven emergency feedwater pump steam supply drain line check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	
EFS-VLV-109C	Turbine driven emergency feedwater pump steam supply drain line check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	
EFS-VLV-109D	Turbine driven emergency feedwater pump steam supply drain line check	Check	Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	

Table 3.9-14 Valve Inservice Test Requirements (Sheet 60 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~NFS~~FWS-VLV-511A	Main feedwater check	Check	Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
~~NFS~~FWS-VLV-511B	Main feedwater check	Check	Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
~~NFS~~FWS-VLV-511C	Main feedwater check	Check	Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
~~NFS~~FWS-VLV-511D	Main feedwater check	Check	Transfer Close Maintain Close	Active	BC	Check Exercise/ Refueling Outage	3
~~NFS~~FWS-SMV-512A	Main feed water isolation	System medium actuated Gate (using valve inside pressure to close)	Maintain Close Transfer Close	Active Active-to-Fail Containment Isolation Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	11

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 61 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~NFS~~FWS-SMV-512B	Main feed water isolation	System medium actuated Gate (using valve inside pressure to close)	Maintain Close Transfer Close	Active Active-to-Fail Containment Isolation Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	11
~~NFS~~FWS-SMV-512C	Main feed water isolation	System medium actuated Gate (using valve inside pressure to close)	Maintain Close Transfer Close	Active Active-to-Fail Containment Isolation Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Cold Shutdown Operability Test	11

Table 3.9-14 Valve Inservice Test Requirements (Sheet 78 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CSS-MOV-004D	Containment spray header containment isolation	Remote MO Gate	Maintain Close Transfer Open Transfer Close	Active Containment Isolation Remote Position Safety Seat Leakage	A	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/Quarterly Operability Test Leak Test/ Refueling Outage	
CSS-VLV-005A	Containment spray header containment isolation	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	~~BC~~ AC	Check Exercise /Refueling Outage	
CSS-VLV-005B	Containment spray header containment isolation	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	~~BC~~ AC	Check Exercise /Refueling Outage	
CSS-VLV-005C	Containment spray header containment isolation	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	~~BC~~ AC	Check Exercise /Refueling Outage	
CSS-VLV-005D	Containment spray header containment isolation	Check	Maintain Close Transfer Open Transfer Close	Active Containment Isolation	~~BC~~ AC	Check Exercise /Refueling Outage	

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 79 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~CSS-VLV-023A~~	~~Containment spray header containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~CSS-VLV-023B~~	~~Containment spray header containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~CSS-VLV-023C~~	~~Containment spray header containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~CSS-VLV-023D~~	~~Containment spray header containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 102 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
NCS-SRV-435B	Reactor coolant pump component cooling water return line relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years and 20% in 4 Years	
NCS-VLV-439A	Reactor coolant pump component cooling water return line check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
NCS-VLV-439B	Reactor coolant pump component cooling water return line check	Check	Maintain Open Transfer Open Transfer Close	Active	BC	Check Exercise/ Refueling Outage	3
~~NCS-VLV-452A~~	~~Reactor coolant pump component cooling water supply containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~NCS-VLV-452B~~	~~Reactor coolant pump component cooling water supply containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 113 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
PSS-MOV-052B	Containment spray/residual heat removal heat exchanger downstream sampling line isolation	Remote MO Globe	Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/ Quarterly Operability Test	
PSS-MOV-052C	Containment spray/residual heat removal heat exchanger downstream sampling line isolation	Remote MO Globe	Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/ Quarterly Operability Test	
PSS-MOV-052D	Containment spray/residual heat removal heat exchanger downstream sampling line isolation	Remote MO Globe	Maintain Close Transfer Close	Active Remote Position	B	Remote Position Indication, Exercise/2 Years Exercise Full Stroke/ Quarterly Operability Test	

Table 3.9-14 Valve Inservice Test Requirements
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Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
PSS-MOV-071	Post accident sampling return line containment isolation	Remote MO Globe	Maintain Close Transfer Close	Active-to-Fail Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/ Quarterly Operability Test	5
PSS-VLV-072	Post accident sampling return line containment isolation	Check	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Containment Isolation Leak Test Check Exercise/Refueling Outage	3 5
~~PSS-VLV-091~~	~~Post accident sampling return line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 122 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RWS-AOV-022	Refueling water storage pit purification return line containment isolation	Remote AO weir type diaphragm	Maintain Close Transfer Close	Active-to-Fail Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/ Quarterly Operability Test	5
RWS-VLV-023	Refueling water storage pit purification return line containment isolation	Check	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	AC	Containment Isolation Leak Test Check Exercise/ Refueling Outage	3 5
RWS-VLV-003	Refueling water storage pit purification line containment isolation check	Check	Maintain Close	Passive Containment Isolation Safety Seat Leakage	A	Containment Isolation Leak Test	5
~~RWS-VLV-073~~	~~Refueling water storage pit purification return line containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 123 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
RWS-VLV-012A	Refueling water recirculation pump discharge check	Check	Transfer Open	Active	BC	Check Exercise /Refueling Outage	
RWS-VLV-012B	Refueling water recirculation pump discharge check	Check	Transfer Open	Active	BC	Check Exercise /Refueling Outage	
DWS-VLV-004	Demineralized water supply containment isolation	Manual	Maintain Close	Passive Containment Isolation Safety Seat Leakage	A	Containment Isolation Leak Test	5
DWS-VLV-005	Demineralized water supply containment isolation check	Check	Maintain Close	Passive Containment Isolation Safety Seat Leakage	AC	Containment Isolation Leak Test	3 5
~~CAS-VLV-004~~	~~Instrument air supply containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements
(Sheet 124 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
CAS-MOV-002	Instrument air supply outside containment isolation	Remote MO Globe	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/Refueling Outage Operability Test	5 6
~~DWS-VLV-006~~	~~Demineralized water supply containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive~~ ~~Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~

Table 3.9-14 Valve Inservice Test Requirements (Sheet 128 of 151)

Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
FSS-VLV-006	FPWSS line to reactor cavity containment isolation check	Check	Maintain Close	Containment Isolation Safety Seat Leakage	AC	Containment Isolation Leak Test	5
~~FSS-VLV-002~~	~~FPWSS line to filter unit containment isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
~~FSS-VLV-005~~	~~FPWSS line to reactor cavity containment isolation test~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation~~ ~~Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
VCS-AOV-304	Containment High Volume Purge Supply Line Containment Isolation Outside of CV	Remote AO Butterfly	Maintain Close Transfer Close	Active-to-Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indication, Exercise/2 Years Containment Isolation Leak Test Exercise Full Stroke/ Cold Shutdown Operability Test	5

Table 3.9-14 Valve Inservice Test Requirements
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Valve Tag Number	Description	Valve/ Actuator Type	Safety-Related Missions	Safety Functions(2)	ASME IST Category	Inservice Testing Type and Frequency	IST Notes
~~RMS-VLV-004~~	~~Containment Air Sampling Return Line Containment Isolation test valve~~	~~Manual~~	~~Maintain Close~~	~~Passive Containment Isolation Safety Seat Leakage~~	~~A~~	~~Containment Isolation Leak Test~~	~~5~~
NCS-SRV-406A	Reactor coolant pump component cooling water outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years	
NCS-SRV-406B	Reactor coolant pump component cooling water outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years	
NCS-SRV-406C	Reactor coolant pump component cooling water outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years	
NCS-SRV-406D	Reactor coolant pump component cooling water outlet relief	Relief	Maintain Close Transfer Open Transfer Close	Active	BC	Class 2/3 Relief Valve Tests/10 Years	

(Reference 3.10-5) as it relates to qualifying equipment to withstand the effects of postulated earthquakes, and the requirements of Appendix B to 10 CFR 50, as it relates to quality assurance criteria, are discussed in Section 3.7 and Chapter 17, respectively.

The seismic qualification and documentation procedures used for safety-related mechanical and electrical equipment and their supports are in accordance with the "IEEE Recommended Practice for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations", ANSI/IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~), as endorsed by the NRC, RG 1.100, Revision 3~~2~~, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Reference 3.10-7).

The US-APWR mechanical and electrical equipment seismic qualification meets IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~) as modified by RG 1.100 (Reference 3.10-7) for qualification by either analysis, testing or by a combination of both testing and analysis~~, and as supplemented with the "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", IEEE Std 344-2004 (Reference 3.10-8) for use to seismically qualify equipment by an experience-based approach. IEEE Std 344-2004 (Reference 3.10-8) is to be endorsed by RG 1.100 (Reference 3.10-7) in a future revision as indicated in "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment", NUREG-0800, SRP 3.10 (Reference 3.10-9). Experience-based qualification is not used for any equipment~~.

The qualification of the design of safety-related, seismic category I mechanical equipment to assure the structural integrity of pressure boundary components follows the guidance provided in the ASME Boiler and Pressure Vessel Code, Section III (Reference 3.10-10). The US-APWR implements an operability program for active valves following the guidance in "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants", ASME QME-1-2007 (Reference 3.10-12) ~~"Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants", RG 1.148 (Reference 3.10-11)~~ as discussed in Subsections 3.9.3 and 3.9.6.

For procured equipment, the design and acceptance criteria for the equipment seismic qualification are required to be specified in the purchase specifications which are part of the purchase order. The applicable level of quality assurance and documentation is also required to be specified. The vendor is required to submit a seismic qualification plan/procedure for review and approval prior to performing the test and/or analysis, as required. Submittal of existing documentation is acceptable if documentation is provided correlating the existing data with the requirements in the purchase order. The vendor is to submit the final qualification documentation, in the form of an equipment seismic qualification report (ESQR), for review and approval prior to acceptance of the equipment. The ESQR is to contain the information identified in Subsection 3.10.4, as required, to confirm that the qualification of the equipment meets the purchase specifications.

3.10.1.2 Performance Requirements for Seismic Qualification

The performance requirements for every item of instrumentation and electrical equipment classified as seismic category I as identified in Appendix 3D and Section 3.11 are provided in the corresponding EQSDSs. An EQSDS is developed for every item of instrumentation and electrical equipment classified as seismic category I. Section 3.11 and Appendix 3D provides the environmental conditions of the electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, to be demonstrated before, during and after a seismic event. The equipment qualification file and EQSDS identify the test response spectrum (TRS) and the Required Response Spectra (RRS) for the seismic qualification. The TRS is required to envelope the RRS for qualification of equipment.

The performance requirements for seismic category I active mechanical components are defined in the corresponding equipment specifications along with the system functional requirements as described in Section 3.2, Section 3.9, and in the sections describing the various systems. Subsection 3.10.2.2 and Section 3.9 discuss additional requirements for active pumps, valves, and dampers and these requirements are included in the EQSDSs contained in the equipment qualification file. For other seismic category I mechanical components, the performance requirements are to maintain structural integrity under seismic and other concurrent applicable loading conditions. The demonstration of meeting the performance requirements is included in the EQSDSs for each mechanical component.

3.10.1.3 Performance Criteria

The qualification of safety-related components to safely withstand seismic loadings in combination with other concurrent dynamic loading effects demonstrates that safety-related seismic category I instrumentation and electrical equipment, and mechanical equipment, including active pumps, valves and dampers, are capable of performing their designated safety-related function(s) under the postulated SSE, as defined in Subsection 3.7.1, in combination with other concurrent loadings. Deformation of supports and structures is acceptable at the SSE levels, provided that their designated safety-related functional performance is not compromised and does not compromise the safety-related function of other equipment.

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

The recommended guidance and requirements in IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~) and RG 1.100 (Reference 3.10-7) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment. The methods and guidance in “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants”, ASME QME-1-2007 (Reference 3.10-12), including Appendix QR-A, with exceptions ~~to be~~ provided in ~~a future revision of~~ RG 1.100 (Reference 3.10-7), are ~~also~~ used for seismic qualification of active mechanical equipment.

The US-APWR seismic category I active mechanical and electrical equipment are seismically qualified in accordance with IEEE standards to safely withstand the SSE effects in combination with other applicable dynamic and static loads.

The design limits, load combinations associated with normal operations, postulated accident, and specified seismic and other transient events, and methods for combining dynamic responses for mechanical equipment are described in Subsection 3.9.3. The dynamic loads considered in testing of instrumentation and electrical equipment, are seismic loads, hydrodynamic, and vibratory loads, as applicable, as discussed in Section 3.11.

Recent seismic research, including recently published attenuation relations, indicates that earthquakes in the central and eastern United States have more energy content in the high-frequency range than earthquakes in the western United States. Therefore, the COL Applicant is to investigate if site-specific in-structure response spectra generated for the COL application may exceed the standard US-APWR design's in-structure response spectra in the high-frequency range. Accordingly, the COL Applicant is to consider the functional performance of vibration-sensitive components, such as relays and other instrument and control devices whose output could be affected by high frequency excitation.

The potential failure modes of the high frequency-sensitive component types and assemblies are considered in order to demonstrate the suitability of the equipment for high-frequency seismic environments. The generic failure modes involving inadvertent change of state, contact chatter, signal change/drift, and connection problems due to high frequency effects are the main focus of the high frequency qualification testing. High frequency failures resulting from improper design of mounting, inadequate design connections and fasteners, mechanical misalignment/binding of parts and the rare case of failure of a component part, will result from the same structural failure modes as those experienced during low frequency content spectra qualification testing in accordance with IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~). Because the safety-related equipment will experience higher stresses and deformations when subjected to the low frequency excitation, these failure modes are more likely to occur under the low frequency testing. Failure modes related to improper mounting, inadequate securing of connections, poor quality joints (cyclic strain effects), etc., are precluded by quality assurance inspection and process/design controls.

Potentially high frequency sensitive components include: electro-mechanical relays; electro-mechanical contactors; circuit breakers; auxiliary contacts; control switches; transfer switches; process switches and sensors; potentiometers; and digital/solid-state devices (mounting and connections only).

Acceptable methods for resolving high frequency concerns not already addressed by certified design qualification where site-specific in-structure response spectra generated for the COL application results in high frequency exceedances of the standard design in-structure response spectra include: review existing equipment qualification test data for adequate high frequency input motion; review circuits containing potentially sensitive items for inappropriate system actions due to intermediacy or set point drifts; or screening test to confirm equipment does not have high frequency vulnerabilities.

If existing test data are not available and a system and control logic review indicates that inadvertent change of state or intermediacy must be considered, then one of the following high frequency screening tests are used to demonstrate lack of sensitivity to high frequency vibrations in the 25-50 Hz range where the function is monitored during the screening test followed by post test functional testing: sine sweep (fast linear rate, traditional log rate); sine beat at 1/6 octave spacing; band-limited white noise; or, random multifrequency time history.

The above testing is not a qualification test but is intended to determine if equipment is potentially sensitive to high-frequency excitation. If the screening tests determine that equipment is potentially sensitive to high-frequency excitation ("screened-in"), then full-scale qualification testing including testing over the range of high-frequency exceedances is required to assure that unacceptable components are not present in the set of qualified certified design equipment and functional systems.

In conjunction with the above, for the purpose of qualification of equipment by analysis, the rigid range is defined as having a natural frequency greater than 50 Hz. For the purpose of testing equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment, to be tested, is sensitive to response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz.

The US-APWR utilizes the following methods for seismic qualification of equipment based on the type, size, shape, and complexity of the equipment configuration, whether the safety function can be assessed in terms of operability or structural integrity alone, and the reliability of the conclusions:

- Predict the equipment's performance by analysis
- Test the equipment under simulated seismic conditions
- Qualify the equipment by a combination of test and analysis

The US-APWR seismic category I equipment is qualified to show that it can perform its safety-related function during and after a postulated earthquake. The seismic qualification considers interfaces and the effects of the amplification within the equipment due to the interfaces and supporting structure. The function of the equipment is dependent on the equipment itself and the system in which it is to function. The safety-related function is determined as that required both during and after a postulated earthquake, which could be different. For example, an electrical device may be required to have no spurious operations during the postulated earthquake or to perform an active function both during, and after, the postulated earthquake, or it may be required to survive during the postulated earthquake and perform an active function after the postulated earthquake, or any combination of these. Another device may only be required to maintain structural integrity during and after the postulated earthquake.

The functionality of mechanical and electrical equipment during and after a postulated earthquake of magnitude up to and including the SSE for static and dynamic loads from normal, Anticipated Operational Occurrence and accident load conditions is assured by tests and/or analyses. The horizontal and vertical SSE RRS curves developed at the damping of interest, as discussed in Subsections 3.7.1 and 3.7.3, form the basis for the seismic qualification of the equipment. The equipment is demonstrated to withstand the equivalent effect of five OBE excitations followed by one SSE for qualification without loss of structural integrity and functionality, as required.

With the elimination of the OBE from design considerations, two alternatives exist that essentially maintain the requirements provided in IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~) to qualify equipment with the equivalent of five OBE events followed by one SSE event (with ten maximum stress cycles per event). Of these alternatives, the equipment is qualified with five 1/2 SSE events followed by one full SSE event (with ten maximum stress cycles per event).

In terms of maximum stress cycles for fatigue analysis, in accordance with SECY-93-087 (Reference 3.10-4), this is equivalent to any of the following:

- 20 cycles of SSE,
- 50 cycles of 1/2 SSE and 10 cycles of SSE,
- 150 cycles of 1/3 SSE and 10 cycles of SSE.

Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five 1/2 SSE events when followed by one full SSE may be used in accordance with ~~Appendix D of IEEE Std 344-1987 (Reference 3.10-6) and~~ Figure D.1 of IEEE Std 344-2004 (Reference 3.10-8).

Selection of damping values for equipment to be qualified is made in accordance with "Damping Values for Seismic Design of Nuclear Power Plants", RG 1.61, Rev.1 (Reference 3.10-13) and IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~). Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism.

Qualification of seismic category I mechanical and electrical equipment by testing is the preferred method for complex equipment which must perform an active function during the SSE. The analysis method alone is not recommended for complex equipment that cannot be modeled to correctly predict its response and functionality. Analysis without testing is acceptable only if structural integrity alone can assure the design-intended function. When complete testing is impractical, then the qualification is performed by a combination of test and analysis.

Equipment previously qualified by means of tests and analyses equivalent to those described herein can be used if proper documentation is provided.

Testing

The seismic qualification testing inputs and methods for qualification of mechanical and electrical equipment are performed in accordance with the guidelines provided in IEEE Std 344-2004~~1987~~, Section 8~~7~~ (Reference 3.10-8~~6~~). Equipment is tested in its operational condition and functionality is verified during and after testing. Loadings for the normal operation of the equipment, such as thermal and flow-induced loads, are simulated and concurrently superimposed upon the seismic and other dynamic loading to the extent practicable. For seismic and dynamic loads, the actual test input is characterized in the same manner as the required input motion to the equipment and the conservatism in amplitude is demonstrated. The TRS envelopes the RRS except for equipment not sensitive to high frequency motion with exceedances in the 25-50 Hz range.

Seismic testing is performed by subjecting equipment to vibratory motion that conservatively simulates that postulated at the equipment mounting location. Factors considered involve the location of the equipment, the nature of the equipment, the nature of the postulated earthquakes, and whether the equipment is to be used in one application or many (proof testing or generic testing). Equipment is conservatively tested considering the multidirectional effects of the postulated earthquakes.

The seismic analysis methods used are performed in accordance with the guidelines in IEEE Std 344-2004~~1987~~, Section 7~~6~~ (Reference 3.10-8~~6~~). Two approaches can be used to seismically qualify equipment by analysis for a number of fatigue inducing smaller earthquake events followed by an SSE using the methods in accordance with IEEE Std 344-2004~~1987~~, Section 7~~6~~ (Reference 3.10-8~~6~~). Qualification by analyses without testing is acceptable if the structural integrity alone can assure the intended design function for the equipment. The two approaches are dynamic analysis and static coefficient analysis. The method utilized is one that takes into account the complexity of the equipment and adequacy of analytical techniques to properly predict the equipment's safety-related functions while under seismic excitation and most accurately represents the equipment's performance under seismic conditions. The method to use is that which most accurately represents the equipment's performance under seismic conditions and is also based on the perceived margin of strength of the equipment.

For dynamic analysis, the equipment and any secondary structural supports are modeled to adequately represent their mass distribution and stiffness characteristics and a modal analysis is performed to determine whether the equipment is rigid or flexible. Rigid equipment can be analyzed using static analysis and the seismic acceleration associated with the mounting location. Flexible equipment can be analyzed using its dynamic response computed from a response spectrum, time-history, or other analysis methods.

When the static coefficient analysis is used, the determination of natural frequencies is not required and the acceleration response of the equipment is assumed to be the maximum peak of the in-structure RRS at 5% damping. Subsection 3.7.3 provides additional discussion on the use of the equivalent static load method of analysis. A static coefficient of 1.5 is used to take into account the effects of multi-frequency excitation and multi-mode response. The increased acceleration values are used as equivalent static load factors applied to the entire mass of the equipment being evaluated. The static coefficient analysis method is used only for the evaluation of structural integrity of equipment. The static analysis method alone is not sufficient for the qualification of safety-related active equipment where the demonstration of operability is required.

When one of the analysis methods described above is used, it is performed with a number of smaller earthquake events that contain a fatigue-inducing potential that is similar to the postulated earthquake response motion at the mounting of the equipment. The number of smaller earthquake events and their fatigue-inducing potential is important only for low-cycle fatigue-sensitive equipment. The analysis will determine that the structural integrity of the equipment is maintained in combination with other applicable loads during the smaller earthquake event. The analysis must show that the smaller fatigue inducing earthquake events followed by an SSE do not result in failure of the equipment to perform its safety-related function. The resulting maximum stresses under applicable loading conditions must be shown to be less than the allowable.

When analyses are used for qualification, the combination of multi-modal and multi-directional responses are made in accordance with "Combining Modal Responses and Spatial Components in Seismic Response Analysis", RG 1.92, Revision 2 (Reference 3.10-16).

Combined Testing and Analysis

The methods used for combined testing and analysis are performed in accordance with the guidelines in IEEE Std 344-2004~~1987~~, Section 9~~8~~ (Reference 3.10-8~~6~~). Combined testing and analysis is utilized when the equipment cannot be practically qualified by analysis or testing alone. Factors used in determining the use of this method include size of the equipment, its complexity, or the large number of similar configurations. Large equipment, such as motors, generators, and multi-bay equipment racks and consoles may be impractical to test at full levels due to limitations in vibration test equipment. Modal testing and analysis can serve as an aid to qualification of large and complex systems. Modal testing is used as the method to determine resonant frequencies, mode shapes, and as a lower bound for modal damping. A modal test may be performed to correlate the frequencies and mode shapes, determined during the analysis, with the measured response of complex system. Extrapolation for similar equipment can be utilized for equipment that was previously qualified and differs only in size or in specific qualified devices located in the assembly or structure.

Interaction of Category II with Seismic Category I Equipment

Seismic category II equipment, as defined in Subsection 3.2.1, is designed and analyzed for the SSE event, using the same methods as specified for seismic category I equipment, to demonstrate structural integrity so as not to collapse on, or adversely interfere with seismic category I equipment. Seismic category I equipment is protected from non-seismic equipment by isolation or the use of barriers when possible. If isolation is not possible, then the equipment is designed and analyzed as seismic category II to maintain structural integrity to withstand an SSE event.

3.10.2.1 Seismic Qualification of Instrumentation and Electrical Equipment

Seismic qualification of seismic category I instrumentation and electrical equipment is demonstrated by either type testing or a combination of test and analysis. The selection of qualification method employed by US-APWR for a particular item of equipment is based upon many factors including: practicability, complexity of equipment, economics, and availability of previous seismic qualification data/reports. The qualification method employed for a particular item of instrumentation or electrical equipment is identified in the individual EQSDS.

Instrumentation described in RG 1.97 (Reference 3.10-1), including associated mountings, are tested under appropriate seismic and dynamic loadings as described in the RG to assure that the instruments continue to monitor plant variables and systems after a seismic event and/or DBA.

3.10.2.1.1 Type Testing

Type testing can be utilized on a sample of equipment representing a generic group that are similar in materials, design and manufacturing. The sample components are in compliance with the manufacturer's quality control system and specifications for production units. The tested equipment is subjected to environmental and operating cycles that simulate the intended service conditions and safety-related functions for which they are to be qualified.

Multi-frequency testing or single-frequency testing is used for seismic category I instrumentation and electrical equipment in accordance with the guidelines in IEEE Std

344-2004~~1987~~ (Reference 3.10-8~~6~~).

Multi-frequency testing is normally used for hard mounted equipment (floor and wall mounted) where a RRS at the equipment mounting location is identified. The test results are provided in the equipment qualification file (and the EQSDS for the individual equipment) and the TRS is shown to envelope the RRS over the entire frequency range of interest, except for equipment not sensitive to high frequency motion with exceedances in the 25-50 Hz range.

Single-frequency testing can be used for line-mounted equipment and other equipment as recommended by IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~) and RG 1.100 (Reference 3.10-7). Required input motion (RIM) in seismic evaluations is normally associated with components in distributions systems (piping and duct) lines where the single mode seismic input to the component is dominated by the seismic response of the distribution system (line) and qualification is performed by generic application to a wide range of line frequencies. For the US-APWR, piping and duct systems are generically designed to limit the peak acceleration experienced by the equipment mounted on them to a value less than the specified RIM acceleration, which is 6.0g horizontal and 6.0g vertical in accordance with “IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants”, IEEE Std 382-1996 (Reference 3.10-17). For line-mounted equipment that is not qualified to the generic level of 6.0g, the seismic input motion is determined from the response of the system analysis in which it is located. The method for qualification of line-mounted equipment is performed in accordance with the guidance in IEEE Std 344-2004~~1987~~, Section 8.6.7~~7.6.7~~ (Reference 3.10-8~~6~~) and IEEE Std 382-1996 (Reference 3.10-17), with justification and test results provided in the equipment qualification file.

3.10.2.1.2 Test and Analysis

The US-APWR utilizes a combination of test and analysis to qualify seismic category I instrumentation and electrical equipment. The test methods utilized are similar to those described above for type testing along with static and/or dynamic analysis. These methods can be used to establish input response requirements at sub-component locations. This approach can be used to justify the extrapolation of tests on a single electrical cabinet, or a small number of connected cabinets, to qualify an assembly. Analysis can be used to: explain unexpected behavior during a test; obtain a better understanding of the dynamic behavior of the equipment so that the proper test can be defined; or obtain a measure of expected response before a test. The documentation is included in the equipment qualification file and the EQSDSs.

3.10.2.2 Seismic and Operability Qualification of Active Mechanical Equipment

The methods and procedures used for qualifying active mechanical equipment (i.e., valves, pumps, and dampers) are described in Section 3.9, Subsection 3.10.2, and this subsection. Analysis, test, or a combination of test and analysis are used for qualification of seismic category I active mechanical equipment to show it maintains structural integrity (including pressure retention), and operability. The methods used assure equipment functionality and operability for its intended safety-related function under required plant conditions.

determined by dynamic analysis.

The procedures acceptable to the NRC for implementing the regulations with respect to the detailed specification of information pertinent to defining the operating requirements for valve assemblies whose safety-related function is to open, close, or regulate fluid flow are discussed in ASME QME-1-2007 ~~RG 1.148~~ (Reference 3.10-12~~11~~). ~~with supplemental information for application of "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard", ANSI N278.1-1975 (Reference 3.10-19).~~ The functional specifications for valves are addressed in Section 3.9.

The equipment specification for safety-related valves includes the valve operating conditions used to evaluate the valve discs, which experience the maximum design line pressure and maximum differential pressure from plant operating, transient, and accident conditions (including pressure and LOCA). Feedwater line valve discs are evaluated for the effect of dynamic loads by considering the effect of dynamic differential pressure. The equivalent differential pressure is developed from a transient analysis that includes system arrangement and valve closing dynamics. The acceptable limits are specified in the ASME Code, Section III (Reference 3.10-10) for Class 1, 2, and 3 valves. An analysis is performed to verify the design adequacy of the disc for the differential pressure and impact energy on the valve disc during a LOCA.

Seismic category I active pressure relief valves are qualified using methods similar to the above requirements for active valves. The end loads applied during testing include the discharge loads. When a relief valve with extended structure is tested to demonstrate operability, a static load equivalent to the seismic load for active valves is applied to the top of the bonnet and the pressure increased until the valve mechanism actuates. The test pressure is applied to the valve inlet to verify that seat leakage is within the limits specified in the equipment specification. The valve is demonstrated to meet the seismic design requirements and functional requirements when successful actuation during testing is demonstrated.

Seismic category I active check valves, due to their simple characteristics, are qualified using standard design or analysis to assure structural integrity and the ability to operate is assured by the design features. In addition to the design considerations, each type of active check valves undergo a stress analysis including applicable SSE loads for critical parts that could affect the operability of the valve, hydrostatic, and seat leakage test. The valve also undergoes in-situ testing and inspection to assure it remains functional.

Pumps

Seismic category I active pumps listed in Table 3.9-7 are constructed in accordance with the ASME Code, Section III (Reference 3.10-10) and are qualified by tests and analysis to verify that their structural and functional requirements are met and the pumps will operate during and after a seismic event. The qualification of the mechanical portions of seismic category I pumps includes the fluid pressure boundary, the suction and discharge nozzles and the shaft and seal retainers and the impeller assembly.

The qualification of seismic category I active pumps is demonstrated by either test, or a combination of test and analysis. The method of qualification is usually determined by the manufacturer and can be as defined above, or existing documented data can be used with adequate justification. The method used must demonstrate that the pump perform their safety-related function.

Dampers

Safety-related active dampers used on ventilation systems to isolate the HVAC areas, such as the control room habitability system, during the seismic events are listed in Table 3.2-2 and Appendix 3D, and are seismically qualified to operate under faulted conditions on demand.

The above methods assure that the active safety-related valves, pumps, and dampers are qualified for operability during a faulted seismic event and assure that they are to perform their safety-related function as required.

3.10.2.3 Pump Motor and Valve Operator Qualification

The seismic category I active pump motor, active valve motor operators, and appurtances vital to operation of the pump and valve are independently qualified for the specified environment as identified in Appendix 3D and Section 3.11, as well as during an SSE seismic event, in accordance with IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~). The seismic qualification is included in the equipment qualification file and EQSDSs for each piece of safety-related equipment.

3.10.2.4 Seismic Qualification of Other Seismic Category I Mechanical Equipment

The seismic qualification of other seismic category I mechanical equipment identified as not active is demonstrated by analysis to maintain structural integrity and pressure retention under applicable loading conditions. The methods utilized are described in Subsections 3.7.3, 3.9.2, and 3.10.2 and conform to the methods described in IEEE Std 344-2004~~1987~~ (Reference 3.10-8~~6~~) and ASME Code, Section III (Reference 3.10-10).

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

The qualification of safety-related seismic category I electrical and mechanical equipment supports is performed by either tests or analyses to assure their structural capability, including anchorage, to withstand seismic excitation characterized by the RRS at the support mounting location.

Electrical equipment and instrumentation supports (including instrument racks, control consoles, cabinets, and panels) are tested with the equipment installed or an equivalent dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. The input motion for the test is determined by the inservice mounting location of the support. The method for testing supports is the same as that described for equipment in Subsection 3.10.2. If the equipment is installed in a non-operational mode for the support test, the response of the support in the test at the equipment mounting location is monitored and characterized as a RRS to be used for functional qualification of the equipment separately, as described in Subsection 3.10.2. The TRS must be shown to envelope the RRS to qualify the support for structural integrity.

If the electrical equipment supports are qualified by analysis using the methods in

3.10-4 Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs. SECY-93-087, United States Regulatory Commission, April 2, 1993.

3.10-5 Earthquake Engineering Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix S, U.S. Nuclear Regulatory Commission, Washington, DC.

3.10-6 Deleted ~~IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987.~~

3.10-7 Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants. Regulatory Guide 1.100, Rev. 3~~2~~, U.S. Nuclear Regulatory Commission, Washington, DC, September 2009.~~June 1988.~~

3.10-8 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. Institute of Electrical and Electronics Engineers (IEEE) Std 344-2004.

3.10-9 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment. NUREG-0800, Standard Review Plan 3.10, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.10-10 Boiler and Pressure Vessel Code. “Section III, Division 1, Nuclear Power Plant Components,” American Society of Mechanical Engineers.

3.10-11 Deleted ~~Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants. Regulatory Guide 1.148, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.~~

3.10-12 Qualification of Active Mechanical Equipment Used in Nuclear Power Plants. American Society of Mechanical Engineers (ASME) QME-1-2007.

3.10-13 Damping Values for Seismic Design of Nuclear Power Plants. Regulatory Guide 1.61, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.10-14 Guidance for Seismic Qualifications of Class 1 Electric Equipment for Nuclear Power Generating Stations. Institute of Electrical and Electronics Engineers (IEEE) Std 344-1971.

3.10-15 IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations. Institute of Electrical and Electronics Engineers (IEEE) Std 323-1974.

3.10-16 Combining Modal Responses and Spatial Components in Seismic Response Analysis. Regulatory Guide 1.92, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, July 2006.

3.10-17 IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants, Institute of Electrical and Electronics Engineers (IEEE) Std 382-1996 (R2004).

3.10-18 Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler Pressure and Vessel Code. ASME Section XI, American Society of Mechanical Engineers.

3.10-19 Deleted ~~Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard. ANSI/ASME N278.1-1975 (Re-designated and Reaffirmed 1992), American National Standards Institute/American Society of Mechanical Engineers.~~

This program is used for the analysis of a behavior, such as water hammer, safety/relief valve discharge etc. by modeling flow volume and flow path.

The pressure and flow rate time-history can be obtained.

- E/PD STRUDL

  E/PD STRUDL (Reference 3.12-22) is a computer program that has the capability to perform the structural analysis of pipe supports in compliance with ASME Code, Section III, Section NF (Reference 3.12-2), and AISC Codes (Reference 3.12-23).

  This computer program is designed to perform analysis of the pipe support structure, including the base plate flexibility per NRC IE Bulletin 79-02 (Reference 3.12-24) as applicable and perform a code stress check of the various components of the support assembly (e.g., structural stock items, welds, anchor bolts, and support vendor components based on data used from vendor's catalog values per vendor's certified design reports).

- STAAD.Pro v8i

  STAAD.Pro is a general purpose structural analysis program. The program is used for steel stress analysis of piping supports.

3.12.4.1.2 Program Validations

Verification tests are to demonstrate the capability of the computer program to produce valid results for test problems encompassing the range of permitted usage defined by the program documentation. Subsection 3.9.1.2 describes the various methods used for computer program validations.

3.12.4.2 Dynamic Piping Model

For dynamic analysis, the piping system is idealized as a three dimensional space frame. The analysis model consists of a sequence of nodes connected by straight pipe elements and curved pipe elements with stiffness properties representing the piping, and other in-line components.

Piping restraints and supports are idealized as zero length springs with appropriate stiffness values for the restrained degrees of freedom.

In the dynamic mathematical model, the distributed mass of the system, including pipe, contents, and insulation weight, is represented as lumped masses located at each node, which is designated as a mass point.

The minimum number of degrees of freedom in the model is to be equal to twice the number of modes with frequencies below a pre-selected cut-off-frequency.

The following formula is used to determine the spacing between two successive mass points and is based on a simply supported beam that would produce a natural frequency equal to a preselected cut-off-frequency. The PIPESTRESS program uses this formula for mass point spacing.

The following subsections provide a description of the various loads considered in the design load combinations. Table 3.12-4 provides the load combinations used in the design of the pipe supports.

3.12.6.3.1 Dead Weight Loads

The loads are based on the dead weight loading case of the associated piping stress analysis and generally include the weight of the piping system and its contents, and any pipe support components attached directly to the pipe (e.g., clamps or integral attachments). In addition, the dead weight of the support components is considered. In Table 3.12-4, dead weight loads are designated by DL.

3.12.6.3.2 Thermal Expansion Loads

Piping analysis may include several thermal expansion loading cases associated with the four service levels. Support loads from these loading conditions are designated as TH_{MTL}, TH_E and TH_F corresponding to the appropriate Level A, B, C, and D service conditions.

3.12.6.3.3 Friction Loads

Friction loads are the result of movement of the pipe across the surface of a support member. Such loads are manually calculated. In Table 3.12-4, friction loads are designated by *F*.

3.12.6.3.4 Wind Loads

Piping exposed to the environment (e.g., yard piping) may be subjected to wind loads. Such piping is analyzed for design basis wind loads. In Table 3.12-4, support loads due to wind are designated by W. All safety-related piping systems are located inside wind protected structures; therefore are not subject to wind and tornado loading.

3.12.6.3.5 Tornado Loads

Piping exposed to the environment (e.g., yard piping) may be subjected to tornado loads. Such piping is analyzed for bounding effects at building penetrations exposed to design basis tornado loads. In Table 3.12-4, support tornado loads due to tornado are designated by W_T. All safety-related piping systems are located inside tornado protected structures; therefore are not subject to tornado and wind loading.

3.12.6.3.6 System Operating Transient Loads

Dynamic loads such as safety/relief valve discharge and water/steam hammer loads are analyzed for piping systems. Support loads from such dynamic analyses are designated as L_{DF} in Table 3.12-4.

3.12.6.3.7 Safe-shutdown Earthquake Loads

Piping is analyzed for both inertial loads and loads due to seismic anchor movements. Piping loads resulting from these loads are designated as safe-shutdown earthquake inertia loads (SSEI) and safe-shutdown earthquake anchor loads (SSEA) in Table 3.12-4.

Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, Standard Review Plan 3.9.2, Rev.3, U.S. Nuclear Regulatory Commission, March 2007.

3.12-17 Piping Benchmark Problems. Vol. 1, NUREG/CR-1677, August 1980, Piping Benchmark Problems. Vol. 2, NUREG/CR-1677, August 1985, U.S. Nuclear Regulatory Commission, Washington, DC.

3.12-18 PIPESTRESS, Piping Stress Analysis Program, Version 3.6.2~~0~~.

3.12-19 Abaqus, Finite Element Structural Analysis Program, Version 6.7, SIMULIA, Providence, RI.

3.12-20 ANSYS, Advanced Analysis Techniques Guide, Release 11.0, ANSYS, Inc., 2007.

3.12-21 RELAP-5, Transient Hydraulic Analysis Program, MOD 3.2, Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID.

3.12-22 E/PD STRUDL, Release 1197, PHI-DELTA Inc., 42 Holbrook Avenue, Braintree. MA 02184, January 23, 1998.

3.12-23 Manual of Steel Construction. American Institute of Steel Construction, 9th Edition, 1989.

3.12-24 Pipe Support Base Plate Designs Using Concrete Expansion Bolts. IE Bulletin No. 79-02, Rev.1, U.S. Nuclear Regulatory Commission, 1979.

3.12-25 Piping Benchmark Problems for the Westinghouse AP600 Standardized Plant. NUREG/CR-6414, August 1996, U.S. Nuclear Regulatory Commission, Washington, DC.

3.12-26 Guidelines for Evaluating Fatigue Analyses incorporating the Life Reduction of Metal Components Due to the Effects of the Light Water Reactor Environment for New Reactors. Regulatory Guide 1.207, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.12-27 Thermal Stresses In Piping Connected To Reactor Coolant Systems. NRC Bulletin No. 88-08 including Supplement 3, U.S. Nuclear Regulatory Commission, Washington, DC, 1988.

3.12-28 Cracking In Feedwater Piping. NRC IE Bulletin No. 79-13, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, 1979.

3.12-29 Pressurizer Surge Line Thermal Stratification. NRC Bulletin No. 88-11, U.S. Nuclear Regulatory Commission, Washington, DC, 1988.

3.12-30 ASME Boiler and Pressure Vessel Code, Section III, Division 1 – Appendices, Nonmandatory Appendix O, 1992 Edition.

3.12-31 Functional Capability of Piping Systems. NUREG-1367, U.S. Nuclear Regulatory Commission, Washington, DC, November 1992.

3.12-32 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, 2001 Edition, The American Society Of Mechanical Engineers.

3.12-33 ASME Boiler and Pressure Vessel Code, Section III, Division 1 – Appendices, Nonmandatory Appendix F, 2001 Edition.

3.12-34 Structural Welding Code – Steel. AWS D1.1/D1.1M, 2006, American Welding Society.

3.12-35 Service Limits and Loading Combinations for Class 1 Linear-Type Supports. Regulatory Guide 1.124, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, February 2007.

3.12-36 Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports. Regulatory Guide 1.130, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.12-37 Code Requirements for Nuclear Safety Related Concrete Structures." ACI-349, American Concrete Institute, 2001.

3.12-38 Anchoring Components and Structural Supports in Concrete. Regulatory Guide 1.199, U.S. Nuclear Regulatory Commission, Washington, DC, November 2003.

3.12-39 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 344-2004, ~~Appendix D,~~ Institute of Electrical and Electronic Engineers Power Engineering Society, New York, New York, June 2005.

3.12-40 Evaluation of Potential for Pipe Breaks, Report of U.S. NRC Piping Review Committee. NUREG-1061, Volume 4, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

Table 3.12-1 ASME Code, Section III, Class 1, 2, 3, CS and Support Load Symbols and Definitions

Load Symbol	Load Definition
DL	Dead Load (The dead weight consists of the weight of the piping, insulation, and other loads permanently imposed upon the piping)
P	Design Pressure
P_R	Range of Service Pressure
P_M	Maximum Service Pressure
F	Friction Loads
TH_{MTL}	ASME Service Level A (Normal) and Service Level B (Upset) Miscellaneous Thermal Loads with Thermal Stratification, Thermal Cycling Effects and static displacement of pre-stressed concrete containment vessel during normal and upset conditions
TH_E	ASME Service Level C (Emergency) Thermal Load
TH_F	ASME Service Level D (Faulted) Thermal Load
TH_{DISCON}	Thermal Discontinuity Loads
TH_{GRAD}	Thermal Radial Gradient Loads
L_{DM}	Design Mechanical Loads
L_{DMS}	Design Mechanical Loads (sustained)
SSEI	Safe-Shutdown Earthquake Inertia Loads
SSEA	Safe-Shutdown Earthquake Anchor Loads
SE	SE is Support self weight excitation, the effect of the acceleration of the support mass caused by building inertial loads such as *SSEI*
SET	Building Settlement
L_{DFN}	ASME Service Level A (Normal) Dynamic Fluid Loads associated with hydraulic transients such as relief/safety valve open or water/steam hammer
L_{DFU}	ASME Service Level B (Upset) Dynamic Fluid Loads associated with hydraulic transients such as relief/safety valve open or water/steam hammer
L_{DFE}	ASME Service Level C (Emergency) Dynamic Fluid Loads associated with hydraulic transients such as relief/safety valve open or water/steam hammer
L_{DFF}	ASME Service Level D (Faulted) Dynamic Fluid Loads associated with hydraulic transients such as relief/safety valve open or water/steam hammer
DBPB	Design Basis Pipe Breaks, include LOCA and non-LOCA, and static displacement of pre-stressed concrete containment vessel during normal and upset conditions
LOCA	Loss-of-Coolant Accident
W	Design Basis Wind Load
W_T	Tornado Wind Load

Table 3.12-2 Loading Combinations for ASME Code, Section III, Class 1 Piping (Sheet 1 of 2)

Condition	Service Level	Category	Loading	Equation (NB-3650)[4]	Stress Limit[4]
Design	-	Primary Stress	P, DL, L_{DM} (including L_{DFN})	Eq. 9 NB-3652	1.5 S_m
Normal /Upset	A/B	Primary + Secondary Stress Intensity Range (SIR)	[3] P_R, TH_{MTL}, TH_{DISCON}, L_{DFN}, L_{DFU}, *SSEI*, *SSEA*	Eq. 10 NB-3653.1	3 S_m
		Peak SIR	P_R, TH_{MTL}, TH_{DISCON}, TH_{GRAD}, L_{DFN}, L_{DFU}, *SSEI*, *SSEA*	Eq. 11 NB-3653.2	
		Thermal Bending SIR	[2] TH_{MTL}	Eq. 12 NB-3653.6(a)	3 S_m
		Primary + Secondary Membrane + Bending SIR	[2] P_R, TH_{DISCON}, L_{DFN}, L_{DFU}, *SSEI*, *SSEA*	Eq. 13 NB-3653.6(b)	3 S_m
		Alternating Stress Intensity (Fatigue)	P_R, TH_{MTL}, TH_{DISCON}, TH_{GRAD}, L_{DFN}, L_{DFU}, *SSEI*, *SSEA*	NB-3653.3 NB-3653.4 NB-3653.5 NB-3653.6(c)	
		Thermal Stress Ratchet	TH_{GRAD} (linear)	NB3653.7	
Upset	B	Permissible Pressure	P_M	NB-3654.1	1.1 *Pa*
		Primary Stress	P_M, DL, L_{DFU}	NB-3654.2	Min(1.8 S_m,1.5 S_y)
Emergency	C	Permissible Pressure	P_M	NB-3655.1	1.5 *Pa*
		Primary Stress	P_M, DL, L_{DFE}	NB-3655.2	Min(2.25 S_m,1.8 S_y)
Faulted	D	Permissible Pressure	P_M	NB-3656(b)	2 Pa
		Primary Stress	P_M, DL, L_{DFF}[1], *SSEI*, *DBPB*[1]	NB-3656(a) NB-3656(b)	Appendix-F or Min(3 S_m,2 S_y)
Faulted	D	Secondary Stress	*SSEA*	[5]	6 S_m[5]

Notes:

1. Dynamic loads are to be combined considering timing and causal relationships. SSE and DBPB is combined using the SRSS.
2. The Thermal and Primary plus Secondary Membrane plus Bending Stress Intensity Ranges (Equations 12 and 13) need only be calculated for those load sets that do not meet the Primary plus Secondary Stress Intensity Range (Equation 10) allowable.
3. The earthquake inertial and anchor movement loads used in the Level B Stress Intensity Range and Alternating Stress calculations (Equations 10, 11, 13 and 14) is taken as 1/3 of the peak SSE inertial and anchor movement loads or as the peak SSE inertial and anchor movement loads. If the earthquake loads are taken as 1/3 of the peak SSE loads then the number of cycles to be considered for earthquake loading is to be 300 as derived in accordance with ~~Appendix D of~~ the Institute of Electrical and Electronic Engineers Standard 344-2004 (Reference 3.12-39) If the earthquake loads are taken as the peak SSE loads then 20 cycles of earthquake loading is considered. Also, see Note 2.
4. ASME Boiler and Pressure Vessel Code, Section III.

Table 3.12-4 Loading Combinations for Piping Supports

Condition	Design Loading Combinations
Level A Service	$DL + L_{DMS} + L_{DFN} + TH_{MTL} + F + SET$
Level B Service	$DL + L_{DMS} + L_{DFU} + TH_{MTL} + W + SET$
Level C Service	$DL + L_{DMS} + L_{DFE} + TH_E\ \cancel{TH_{MTL}} + W_T + SET$
Level D Service	$DL + L_{DMS} + L_{DFF} + TH_F\ \cancel{TH_{MTL}} + SET$
	$DL + L_{DMS} + L_{DFF} + TH_{MTL} + DBPB + SET$
	$DL + L_{DMS} + L_{DFF} + TH_{MTL} + SET + SRSS(DBPB + (SSEI + SSEA + SE))$ (1), (2), (3)

Notes:

1. SRSS
2. Dynamic loads are combined by the *SRSS.*
3. Combine *SSEI, SSEA*, and *SE* by absolute sum method. *SE* is support self weight excitation, the effect of the acceleration of the support mass caused by building inertial loads such as *SSEI.*
4. If, during operation, the system normally carries a medium other than water (air, gas, steam), sustained loads should be checked for weight loads during hydrostatic testing as well as normal operation weight loads.

Table 3D-2 US-APWR Environmental Qualification Equipment List
(Sheet 15 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
24	ICT-TE-022	Core Exit Temperature	PCCV	1-3	PAM	4mos	Harsh	Harsh	No (1)	E	I	
25	ICT-TE-023	Core Exit Temperature	PCCV	1-3	PAM	4mos	Harsh	Harsh	No (1)	E	I	
26	ICT-TE-024	Core Exit Temperature	PCCV	1-3	PAM	4mos	Harsh	Harsh	No (1)	E	I	
27	ICT-TE-025	Core Exit Temperature	PCCV	1-3	PAM	4mos	Harsh	Harsh	No (1)	E	I	
28	ICT-TE-026	Core Exit Temperature	PCCV	1-3	PAM	4mos	Harsh	Harsh	No (1)	E	I	
Instruments (Radiation Monitors)												
1	RMS-RE-083A	Main Control Room Outside Air Intake Particulate Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
2	RMS-RE-083B	Main Control Room Outside Air Intake Particulate Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
3	RMS-RE-084A	Main Control Room Outside Air Intake Gas Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
4	RMS-RE-084B	Main Control Room Outside Air Intake Gas Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
5	RMS-RE-085A	Main Control Room Outside Air Intake Iodine Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
6	RMS-RE-085B	Main Control Room Outside Air Intake Iodine Radiation	R/B	14	ESF	5min	Mild	Mild	No (1)	E	I	
7	RMS-RE-091A	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
8	RMS-RE-091B	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
~~8~~9	RMS-RE-092A	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
10	RMS-RE-092B	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
~~9~~11	RMS-RE-093A	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
12	RMS-RE-093B	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
~~10~~13	RMS-RE-094A	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
14	RMS-RE-094B	Containment High Range Area Radiation	PCCV	1-6	ESF PAM	5min, 4mos	Harsh	Harsh	No (1)	E	I	
~~11~~15	RMS-RE-40	Containment Airborne Particulate Radiation	R/B	13-3	Other	36hr*	Mild	Harsh	No (1)	E	I	*Not Required Post Accident

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
22	CVS-AOV-192C	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
23	CVS-AOV-192D	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
24	CVS-LCV-031B	Level Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
25	CVS-LCV-031C	Level Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
26	CVS-LCV-031D	Level Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
27	CVS-LCV-031E	Level Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
28	CVS-AOV-165	Air Operated Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
29	CVS-FCV-048	Flow Control Valve	R/B	7	PB	2wks	Mild	Harsh	No (1)	M	I	
30	CVS-FCV-050	Flow Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
31	CVS-FCV128	Flow Control Valve	R/B	13-3	Other	2wks	Mild	Mild	No (1)	M	I	
32	CVS-FCV-129	Flow Control Valve	R/B	13-3	Other	2wks	Mild	Mild	No (1)	M	I	
33	CVS-SRV-002	Safety Valve	PCCV	1-4	ESF	1yr	Harsh	Harsh	No (1)	M	I	
34	CVS-SRV-201	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
35	CVS-AOV-001A	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
36	CVS-AOV-001B	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
37	CVS-AOV-001C	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
38	CVS-HCV-012	Air Operated Valve	PCCV	1-5	PB	1yr	Harsh	Harsh	No (1)	M	I	
39	CVS-HCV-100	Air Operated Valve	PCCV	1-5	PB	1yr	Harsh	Harsh	No (1)	M	I	
40	CVS-AOV-224	Air Operated Valve	PCCV	1-5	PB	1yr	Harsh	Harsh	No (1)	M	I	
41	CVS-LCV-031F	~~Air Operated~~ Level Control Valve	R/B	7	Other	2 wks	Mild	Harsh	No (1)	M	I	
42	CVS-LCV-031G	~~Air Operated~~ Level Control Valve	R/B	7	Other	2wks	Mild	Harsh	No (1)	M	I	
43	CVS-AOV-196A	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
44	CVS-AOV-196B	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
45	CVS-AOV-196C	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	
46	CVS-AOV-196D	Air Operated Valve	PCCV	1-4	PB	1yr	Harsh	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
40	RHS-SRV-023B	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
41	RHS-SRV-023C	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
42	RHS-SRV-023D	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
Equipment (Emergency Feedwater System)												
1	EFS-MPP-001A	A-Emergency Feedwater Pump	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
2	EFS-MPP-001B	B-Emergency Feedwater Pump	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
3	EFS-MPP-001C	C-Emergency Feedwater Pump	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
4	EFS-MPP-001D	D-Emergency Feedwater Pump	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
5	EFS-MPT-001A	A-Emergency Feedwater Pit	R/B	14	ESF	2wks	Mild	Mild	No (1)	M	I	
6	EFS-MPT-001B	B- Emergency Feedwater Pit	R/B	14	ESF	2wks	Mild	Mild	No (1)	M	I	
7	EFS-MOV-014A	Motor Operated Valve	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
8	EFS-MOV-014B	Motor Operated Valve	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
9	EFS-MOV-014C	Motor Operated Valve	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
10	EFS-MOV-014D	Motor Operated Valve	R/B	8	ESF	2wks	Mild	Harsh	No (1)	M	I	
11	EFS-MOV-017A	A-Emergency Feedwater Control Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
12	EFS-MOV-017B	B-Emergency Feedwater Control Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
13	EFS-MOV-017C	C-Emergency Feedwater Control Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
14	EFS-MOV-017D	D-Emergency Feedwater Control Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
15	EFS-MOV-019A	A-Emergency Feedwater Isolation Valve	R/B	10	ESF	2wks	~~Mild~~ Harsh	Harsh	No (1)	M	I	
16	EFS-MOV-019B	B-Emergency Feedwater Isolation Valve	R/B	10	ESF	2wks	~~Mild~~ Harsh	Harsh	No (1)	M	I	
17	EFS-MOV-019C	C-Emergency Feedwater Isolation Valve	R/B	10	ESF	2wks	~~Mild~~ Harsh	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
18	EFS-MOV-019D	D-Emergency Feedwater Isolation Valve	R/B	10	ESF	2wks	~~Mild~~ Harsh	Harsh	No (1)	M	I	
19	EFS-MOV-101A	A-Emergency Feedwater Pump A-Main Steam Line Steam Isolation Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
20	EFS-MOV-101B	A-Emergency Feedwater Pump B-Main Steam Line Steam Isolation Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
21	EFS-MOV-101C	D-Emergency Feedwater Pump C-Main Steam Line Steam Isolation Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
22	EFS-MOV-101D	D-Emergency Feedwater Pump D-Main Steam Line Steam Isolation Valve	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
23	EFS-MOV-103A, EFS-MOV-103B	A-Emergency Feedwater Pump Actuation Valve on A-steam supply line, A-Emergency Feedwater Pump Actuation Valve on B-steam supply line	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
24	EFS-MOV-103C, EFS-MOV-103D	~~B~~D-Emergency Feedwater Pump Actuation Valve on C-steam supply line, D-Emergency Feedwater Pump Actuation Valve on D-steam supply line	R/B	10	ESF	2wks	Harsh	Harsh	No (1)	M	I	
Equipment (Main Feedwater System)												
1	FWS-SMV~~VLV~~-512A	A-Main Feedwater Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (2)	M	I	
2	FWS-SMV~~VLV~~-512B	B-Main Feedwater Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (2)	M	I	
3	FWS-SMV~~VLV~~-512C	C-Main Feedwater Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (2)	M	I	
4	FWS-SMV~~VLV~~-512D	D-Main Feedwater Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (2)	M	I	
5	FWS-MOV-514A	Motor Operated Valve	R/B	10	PB	2wks	Harsh	Harsh	No (2)	M	I	
6	FWS-MOV-514B	Motor Operated Valve	R/B	10	PB	2wks	Harsh	Harsh	No (2)	M	I	
7	FWS-MOV-514C	Motor Operated Valve	R/B	10	PB	2wks	Harsh	Harsh	No (2)	M	I	
8	FWS-MOV-514D	Motor Operated Valve	R/B	10	PB	2wks	Harsh	Harsh	No (2)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List (Sheet 38 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
33	MSS-AOV-515A	A-Main Steam Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
34	MSS-AOV-515B	B-Main Steam Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
35	MSS-AOV-515C	C-Main Steam Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
36	MSS-AOV-515D	D-Main Steam Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
37	MSS-HCV-565	A-Main Steam Bypass Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
38	MSS-HCV-575	B-Main Steam Bypass Isolation Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
39	MSS-HCV-585	C-Main Steam Bypass Isolation Valve Hand Control Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
40	MSS-HCV-595	D-Main Steam Bypass Isolation Valve Hand Control Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
41	MSS-PCV-515	A-Main Steam Relief Valve	R/B	10	PB	1yr	Harsh	Harsh	No (1)	M	I	
42	MSS-PCV-525	B-Main Steam Relief Valve	R/B	10	PB	1yr	Harsh	Harsh	No (1)	M	I	
43	MSS-PCV-535	C-Main Steam Relief Valve	R/B	10	PB	1yr	Harsh	Harsh	No (1)	M	I	
44	MSS-PCV-545	D-Main Steam Relief Valve	R/B	10	PB	1yr	Harsh	Harsh	No (1)	M	I	
45	MSS-TCV-550A	A-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
46	MSS-TCV-550B	B-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
47	MSS-TCV-550C	C-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
48	MSS-TCV-550D	D-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
49	MSS-TCV-550E	E-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
50	MSS-TCV-550F	F-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
51	MSS-TCV-550G	G-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
52	MSS-TCV-550H	H-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
53	MSS-TCV-550J	J-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
54	MSS-TCV-550K	K-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
55	MSS-TCV-550L	L-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	

Table 3D-2 US-APWR Environmental Qualification Equipment List
(Sheet 39 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
56	MSS-TCV-550M	M-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
57	MSS-TCV-550N	N-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
58	MSS-TCV-550P	P-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
59	MSS-TCV-550Q	Q-Turbine Bypass Valve	T/B	14	~~ESF~~ Other	5min	Mild	Mild	No (3)	M	Non	
60	MSS-MOV-701A	A-Main Steam Drain Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
61	MSS-MOV-701B	B-Main Steam Drain Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
62	MSS-MOV-701C	C-Main Steam Drain Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
63	MSS-MOV-701D	D-Main Steam Drain Isolation Valve	R/B	10	ESF	1yr	Harsh	Harsh	No (1)	M	I	
Equipment (Containment Spray System)												
64	Deleted											
65	CSS-MOV-004A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
66	CSS-MOV-004B	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
67	CSS-MOV-001A	Motor Operated Valve	R/B	6	ESF	1yr	Harsh	Harsh	No (1)	M	I	
68	CSS-MOV-001B	Motor Operated Valve	R/B	6	ESF	1yr	Harsh	Harsh	No (1)	M	I	
69	CSS-MOV-004C	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
70	CSS-MOV-004D	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
71	CSS-MOV-001C	Motor Operated Valve	R/B	6	ESF	1yr	Harsh	Harsh	No (1)	M	I	
72	CSS-MOV-001D	Motor Operated Valve	R/B	6	ESF	1yr	Harsh	Harsh	No (1)	M	I	
73	CSS-MOV-011	Motor Operated Valve	R/B	6	PB	1yr	Mild	Harsh	No (1)	M	I	
Equipment (Component Cooling Water System)												
1	NCS-MPP-001A	A-Component Cooling Water Pump	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
2	NCS-MPP-001B	B-Component Cooling Water Pump	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
(Sheet 41 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[1]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
24	NCS-MOV-020D	Motor Operated Valve	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
25	NCS-RCV-056B	Radiation Control Valve	R/B	8	PB	1yr	Mild	Harsh	No (1)	M	I	
26	NCS-LCV-020	Level Control Valve	R/B	8	PB	1yr	Mild	Harsh	No (2)	M	I	
27	NCS-MOV-145A	Motor Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
28	NCS-MOV-436B	Motor Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
29	NCS-MOV-438A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
30	NCS-MOV-145B	Motor Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
31	NCS-MOV-145C	Motor Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
32	NCS-MOV-145D	Motor Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
33	NCS-MOV-232A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
34	NCS-MOV-232B	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
35	NCS-MOV-233A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
36	NCS-MOV-233B	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
37	NCS-MOV-234A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
38	NCS-MOV-234B	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
39	NCS-MOV-316A	Motor Operated Valve	R/B	~~7~~ 13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
40	NCS-MOV-316B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
41	NCS-SRV-406A	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
42	NCS-SRV-406B	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
43	NCS-SRV-406C	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
44	NCS-SRV-406D	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
45	NCS-SRV-435A	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
46	NCS-SRV-435B	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
47	NCS-MOV-436A	Motor Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
72	NCS-FCV-130B	Flow Control Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
73	NCS-MOV-438B	Motor Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
74	NCS-FCV-132A	Flow Control Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
75	NCS-FCV-132B	Flow Control Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
76	NCS-SRV-513	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
77	NCS-SRV-533	Safety Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
78	NCS-AOV-601	Air Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
79	NCS-AOV-602	Air Operated Valve	R/B	13-3	ESF	1yr	Mild	Harsh	No (1)	M	I	
80	NCS-AOV-661A	Air Operated Valve	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
81	NCS-AOV-662A	Air Operated Valve	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
82	NCS-AOV-661B	Air Operated Valve	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
83	NCS-AOV-662B	Air Operated Valve	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
84	NCS-PCV-012	Pressure Control Valve	R/B	8	PB	1yr	Mild	Harsh	No (1)	M	I	
85	NCS-PCV-022	Pressure Control Valve	R/B	8	PB	1yr	Mild	Harsh	No (1)	M	I	
86	NCS-MOV-321A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
87	NCS-MOV-321B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
88	NCS-MOV-322A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
89	NCS-MOV-322B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
90	NCS-MOV-323A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
91	NCS-MOV-323B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
92	NCS-MOV-324A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
93	NCS-MOV-324B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
94	NCS-MOV-325A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
95	NCS-MOV-325B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
96	NCS-MOV-326A	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	
97	NCS-MOV-326B	Motor Operated Valve	R/B	13-3	PB	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
Equipment (Spent Fuel Pit Cooling and Purification System)												
1	SFP-MPP-001A	A-Spent Fuel Pit Pump	R/B	6	ESF	~~2wks~~ 1yr	Mild	Harsh	No (1)	M	I	
2	SFP-MPP-001B	B-Spent Fuel Pit Pump	R/B	6	ESF	~~2wks~~ 1yr	Mild	Harsh	No (1)	M	I	
3	SFP-MHX-001A	A-Spent Fuel Pit Heat Exchanger	R/B	6	ESF	~~2wks~~ 1yr	Mild	Harsh	No (1)	M	I	
4	SFP-MHX-001B	B-Spent Fuel Pit Heat Exchanger	R/B	6	ESF	~~2wks~~ 1yr	Mild	Harsh	No (1)	M	I	
Equipment (Essential Service Water System)												
1	EWS-MPP-001A	A-Essential Service Water Pump	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
2	EWS-MPP-001B	B-Essential Service Water Pump	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
3	EWS-MPP-001C	C-Essential Service Water Pump	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
4	EWS-MPP-001D	D-Essential Service Water Pump	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
5	EWS-SST-001A	A-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
6	EWS-SST-002A	A-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
7	EWS-SST-001B	B-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
8	EWS-SST-002B	B-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
9	EWS-SST-001C	C-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
10	EWS-SST-002C	C-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
11	EWS-SST-001D	D-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
12	EWS-SST-002D	D-Essential Service Water Pump Outlet Strainer	UHSRS	-	ESF	1yr	Mild	-	-	M	I	
13	EWS-SST-003A	A-Component Cooling Water Heat Exchanger Inlet Strainer	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
14	EWS-SST-003B	B-Component Cooling Water Heat Exchanger Inlet Strainer	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
15	EWS-SST-003C	C-Component Cooling Water Heat Exchanger Inlet Strainer	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
16	EWS-SST-003D	D-Component Cooling Water Heat Exchanger Inlet Strainer	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
17	EWS-MOV-503A	Motor Operated Valve	UHSRS	-	ESF	1yr	Mild	-	-	M	I	

**Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
24	LMS-AOV-053	Air Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
25	LMS-AOV-055	Air Operated Valve	PCCV	1-5	ESF	5min	Harsh	Harsh	No (1)	M	I	
26	LMS-AOV-056	Air Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
27	LMS-AOV-060	Air Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
28	LMS-LCV-010A	Air Operated Valve	PCCV	1-5	ESF	5min	Harsh	Harsh	No (1)	M	I	
29	LMS-LCV-010B	Air Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
30	LMS-AOV-104	Air Operated Valve	PCCV	1-5	ESF	5min	Harsh	Harsh	No (1)	M	I	
31	LMS-AOV-105	Air Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
Equipment (Solid Radioactive Waste Management System)												
1	SMS-MTK-001A	A-Spent Resin Storage Tank	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
2	SMS-MTK-001B	B-Spent Resin Storage Tank	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
3	SMS-AOV-023A	Air Operated Valve	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
4	SMS-AOV-023B	Air Operated Valve	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
5	SMS-AOV-032A	Air Operated Valve	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
6	SMS-AOV-032B	Air Operated Valve	A/B	13-1	Other	1yr	Mild	Harsh	No (3)	M	Non	
Equipment (Process and Post Accident Sampling System)												
1	PSS-AOV-003	Air Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
2	PSS-MOV-006	Motor Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
3	PSS-MOV-013	Motor Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
4	PSS-MOV-023	Motor Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
5	PSS-MOV-031A	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
6	PSS-MOV-031B	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
7	PSS-MOV-052A	Motor Operated Valve	R/B	13-2	Other	1yr	Mild	Harsh	No (1)	M	I	
8	PSS-MOV-052B	Motor Operated Valve	R/B	13-2	Other	1yr	Mild	Harsh	No (1)	M	I	
9	PSS-MOV-052C	Motor Operated Valve	R/B	13-2	Other	1yr	Mild	Harsh	No (1)	M	I	
10	PSS-MOV-052D	Motor Operated Valve	R/B	13-2	Other	1yr	Mild	Harsh	No (1)	M	I	

**Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
11~~9~~	PSS-AOV-062A	Air Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
12~~10~~	PSS-AOV-062B	Air Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
13~~11~~	PSS-AOV-062C	Air Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
14~~12~~	PSS-AOV-062D	Air Operated Valve	PCCV	1-5	ESF	1yr	Harsh	Harsh	No (1)	M	I	
15~~13~~	PSS-AOV-063	Air Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
16~~14~~	PSS-MOV-071	Motor Operated Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
17~~15~~	PSS-MOV-301	Motor Operated Valve	R/B	6	Other	1yr	Mild	Harsh	No (1)	M	~~Non~~ I	
18~~16~~	PSS-MOV-312	Motor Operated Valve	R/B	6	Other	1yr	Mild	Harsh	No (1)	M	~~Non~~ I	
Equipment (Steam Generator Blowdown System)												
1	SGS-AOV-001A	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
2	SGS-AOV-001B	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
3	SGS-AOV-001C	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
4	SGS-AOV-001D	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
5	SGS-AOV-031A	Air Operated Valve	R/B	6	ESF	5min	Harsh	Harsh	No (1)	M	I	
6	SGS-AOV-031B	Air Operated Valve	R/B	6	ESF	5min	Harsh	Harsh	No (1)	M	I	
7	SGS-AOV-031C	Air Operated Valve	R/B	6	ESF	5min	Harsh	Harsh	No (1)	M	I	
8	SGS-AOV-031D	Air Operated Valve	R/B	6	ESF	5min	Harsh	Harsh	No (1)	M	I	
9	SGV-AOV-002A	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
10	SGV-AOV-002B	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
11	SGV-AOV-002C	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
12	SGV-AOV-002D	Air Operated Valve	R/B	10	ESF	5min	Harsh	Harsh	No (1)	M	I	
Equipment (Refueling Water Storage System)												
1	Deleted											
2	RWS-MOV-002	Motor Operated Valve	PCCV	1-4	ESF	5min	Harsh	Harsh	No (1)	M	I	

**Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
10	VRS-MCL-101B	B-Main Control Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
11	VRS-MCL-101C	C-Main Control Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
12	VRS-MCL-101D	D-Main Control Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
13	VRS-MEH-101A	A-Main Control Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
14	VRS-MEH-101B	B-Main Control Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
15	VRS-MEH-101C	C-Main Control Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
16	VRS-MEH-101D	D-Main Control Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
17	VRS-MFU-111A	A-Main Control Room Emergency Filtration Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
18	VRS-MFU-111B	B-Main Control Room Emergency Filtration Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
19	VRS-MFN-111A	A-Main Control Room Emergency Filtration Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
20	VRS-MFN-111B	B-Main Control Room Emergency Filtration Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
21	VRS-MEH-111A	A-Main Control Room Emergency Filtration Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
22	VRS-MEH-111B	B-Main Control Room Emergency Filtration Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
23	VRS-EHD-101A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
24	VRS-EHD-101B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
25	VRS-EHD-102A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
26	VRS-EHD-102B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	14	ESF	1yr	Mild	Mild	No (1)	M	I	
27	VRS-AOD-103A	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
28	VRS-AOD-103B	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
29	VRS-EHD-104A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
30	VRS-EHD-104B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
31	VRS-EHD-105A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
32	VRS-EHD-105B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
33	VRS-EHD-105C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
34	VRS-EHD-105D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
35	VRS-EHD-106A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
36	VRS-EHD-106B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
37	VRS-EHD-106C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
38	VRS-EHD-106D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
39	VRS-EHD-107A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
40	VRS-EHD-107B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
41	VRS-MOD-111A	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
42	VRS-MOD-111B	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
43	VRS-MOD-112A	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
44	VRS-MOD-112B	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
45	VRS-MOD-113A	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
46	VRS-MOD-113B	Motor Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
47	VRS-AOD-121	Air Operated Damper	R/B	2	ESF	5min	Mild	Mild	No (1)	M	I	
48	VRS-AOD-122	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
49	VRS-AOD-131	Air Operated Damper	R/B	2	ESF	5min	Mild	Mild	No (1)	M	I	
50	VRS-AOD-132	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
(Sheet 50 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
Equipment (Engineered Safety Features Ventilation System)												
1	VRS-MFU-001A	A-Annulus Emergency Exhaust Filtration Unit	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
2	VRS-MFU-001B	B-Annulus Emergency Exhaust Filtration Unit	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
3	VRS-MFN-001A	A-Annulus Emergency Exhaust Filtration Unit Fan	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
4	VRS-MFN-001B	B-Annulus Emergency Exhaust Filtration Unit Fan	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
5	VRS-EHD-001A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
6	VRS-EHD-001B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
7	VRS-EHD-002A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
8	VRS-EHD-002B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
9	VRS-EHD-003A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
10	VRS-EHD-003B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
11	VRS-MAH-201A	A-Class 1E Electrical Room Air Handling Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
12	VRS-MAH-201B	B-Class 1E Electrical Room Air Handling Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
13	VRS-MAH-201C	C-Class 1E Electrical Room Air Handling Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
14	VRS-MAH-201D	D-Class 1E Electrical Room Air Handling Unit	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
15	VRS-MFN-201A	A-Class 1E Electrical Room Air Handling Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
16	VRS-MFN-201B	B-Class 1E Electrical Room Air Handling Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
17	VRS-MFN-201C	C-Class 1E Electrical Room Air Handling Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
18	VRS-MFN-201D	D-Class 1E Electrical Room Air Handling Unit Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
19	VRS-MFN-202A	A-Class 1E Electrical Room Return Air Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
20	VRS-MFN-202B	B-Class 1E Electrical Room Return Air Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
21	VRS-MFN-202C	C-Class 1E Electrical Room Return Air Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
22	VRS-MFN-202D	D-Class 1E Electrical Room Return Air Fan	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
23	VRS-MCL-201A	A-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
24	VRS-MCL-201B	B-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
25	VRS-MCL-201C	C-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
26	VRS-MCL-201D	D-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
27	VRS-MEH-201A	A-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
28	VRS-MEH-201B	B-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
29	VRS-MEH-201C	C-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
30	VRS-MEH-201D	D-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
31	VRS-MEH-202A	A-Class 1E I&C Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
32	VRS-MEH-202B	B-Class 1E I&C Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
33	VRS-MEH-202C	C-Class 1E I&C Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
34	VRS-MEH-202D	D-Class 1E I&C Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
35	VRS-MEH-203A	A - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
36	VRS-MEH-203B	B - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
37	VRS-MEH-203C	C - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
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Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
38	VRS-MEH-203D	D - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
39	VRS-MEH-211A	A - Remote Shutdown Console Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
40	VRS-MEH-211B	B - Remote Shutdown Console Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
41	VRS-MEH-204A	A - Class 1E Battery Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
42	VRS-MEH-204B	B - Class 1E Battery Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
43	VRS-MEH-204C	C - Class 1E Battery Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
44	VRS-MEH-204D	D - Class 1E Battery Room In-Duct Heater	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
45~~31~~	VRS-EHD-201A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
46~~32~~	VRS-EHD-201B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
47~~33~~	VRS-EHD-201C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
48~~34~~	VRS-EHD-201D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
49~~35~~	VRS-EHD-202A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
50~~36~~	VRS-EHD-202B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
51~~37~~	VRS-EHD-202C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
52~~38~~	VRS-EHD-202D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
53~~39~~	VRS-EHD-203A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
54~~40~~	VRS-EHD-203B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
55~~41~~	VRS-EHD-203C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
56~~42~~	VRS-EHD-203D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
57~~43~~	VRS-EHD-204A	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List (Sheet 53 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other[(1)]		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
58~~44~~	VRS-EHD-204B	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
59~~45~~	VRS-EHD-204C	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
60~~46~~	VRS-EHD-204D	Electro Hydraulic ~~Motor~~ Operated Damper	R/B	8	ESF	1yr	Mild	Harsh	No (1)	M	I	
61~~47~~	VRS-AOD-205A	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
62~~48~~	VRS-AOD-205B	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
63~~49~~	VRS-AOD-205C	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
64~~50~~	VRS-AOD-205D	Air Operated Damper	R/B	8	ESF	5min	Mild	Mild	No (1)	M	I	
65~~51~~	VRS-MFN-251A	A-Class 1E Battery Room Exhaust Fan	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
66~~52~~	VRS-MFN-251B	B-Class 1E Battery Room Exhaust Fan	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
67~~53~~	VRS-MFN-251C	C-Class 1E Battery Room Exhaust Fan	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
68~~54~~	VRS-MFN-251D	D-Class 1E Battery Room Exhaust Fan	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
69~~55~~	VRS-EHD-251A	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
70~~56~~	VRS-EHD-251B	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
71~~57~~	VRS-EHD-251C	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
72~~58~~	VRS-EHD-251D	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
73~~59~~	VRS-EHD-252A	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
74~~60~~	VRS-EHD-252B	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
75~~61~~	VRS-EHD-252C	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
76~~62~~	VRS-EHD-252D	Electro Hydraulic ~~Motor~~ Operated Damper	PS/B	11	ESF	1yr	Mild	Mild	No (1)	M	I	
77~~63~~	VRS-MAH-301A	A-Safeguard Component Area Air Handling Unit	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
78~~64~~	VRS-MAH-301B	B-Safeguard Component Area Air Handling Unit	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List (Sheet 63 of 64)

Item Num	Equipment Tag	Description	Location		Purpose	Operational Duration	Environmental Conditions	Radiation Condition	Influence of Submergence for Total Integrated Dose	Qualification Process	Seismic Category	Comments
			Building	Zone	RT, ESF, PAM, Pressure Boundary (PB), Other(1)		Harsh or Mild	Harsh or Mild	Yes/No	E=Electrical M=Mechanical	I, II, Non	
42	VWS-TMV-602B	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
43	VWS-TMV-612A	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
44	VWS-TMV-612B	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
45	VWS-TMV-622	Chilled Water Control Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
46	VWS-TMV-632	Chilled Water Control Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
47	VWS-TMV-642	Chilled Water Control Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
48	VWS-TMV-652	Chilled Water Control Valve	R/B	6	ESF	1yr	Mild	Harsh	No (1)	M	I	
49	VWS-TMV-662A	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
50	VWS-TMV-662B	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
51	VWS-TMV-672A	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
52	VWS-TMV-672B	Chilled Water Control Valve	R/B	7	ESF	1yr	Mild	Harsh	No (1)	M	I	
53	VWS-MOV-403	Motor Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
54	VWS-MOV-407	Motor Operated Valve	R/B	6	ESF	5min	Mild	Harsh	No (1)	M	I	
55	VWS-MOV-422	Motor Operated Valve	PCCV	1-6	ESF	5min	~~Mild~~ Harsh	Harsh	No (1)	M	I	
56	VWS-SRV-253A	Safety Valve	PS/B	9	ESF	1yr	Mild	Mild	No (1)	M	I	
57	VWS-SRV-253B	Safety Valve	PS/B	9	ESF	1yr	Mild	Mild	No (1)	M	I	
58	VWS-SRV-253C	Safety Valve	PS/B	9	ESF	1yr	Mild	Mild	No (1)	M	I	
59	VWS-SRV-253D	Safety Valve	PS/B	9	ESF	1yr	Mild	Mild	No (1)	M	I	

Table 3D-2 US-APWR Environmental Qualification Equipment List
(Sheet 64 of 64)

1. Identification number for “Influence of Submergence for Total Integrated Dose”
   (1) Components with no possibility of submergence.
   (2) These components can be submerged in case of HELB, however these components are not required to assure the safety function (including components with alternativeness).
   (3) Non-safety related components.
2. Identification number for “Purpose”
   (1) All active valves in Table 3D-2 have the function and operating duration of “PB-1yr” in addition to any other requirements.

CONTENTS

~~TABLES~~

~~FIGURES~~

ACRONYMS AND ABBREVIATIONS

~~ASCE~~	~~American Society of Civil Engineers~~
~~FE~~	~~finite element~~
~~ISRS~~	~~in-structure response spectra~~
PCCV	prestressed concrete containment vessel
R/B	reactor building
~~RCL~~	~~reactor coolant loop~~
~~SSI~~	~~soil-structure interaction~~
~~SRSS~~	~~square root sum of the squares~~

3H Model Properties for Lumped Mass Stick Models of R/B-PCCV-Containment Internal Structure on a Common Basemat

3H.1 Introduction

~~This Appendix discusses the properties of the reactor building (R/B), prestressed concrete containment vessel (PCCV), and containment internal structure lumped mass stick models used for the seismic analysis. This Appendix also discusses the properties of the east and west power source buildings (PS/Bs) lumped mass stick model used for the seismic analysis.~~

~~This Appendix presents the results of the seismic analyses performed on the lumped mass stick models, but does not address seismic analysis methods, procedures used for analytical modeling, soil-structure interaction (SSI) analysis methods, development of in-structure response spectra (ISRS), components of earthquake motion, combination of modal responses, interaction of non-seismic Category I structures with seismic Category I structures, effects of parameter variations on ISRS, use of constant vertical static factors, methods used to account for torsional effects, comparison of responses (time history versus response spectra analysis results), methods of seismic analysis of dams, determination of dynamic stability of seismic Category I structures, or analysis procedures for damping. Those topics are addressed in Subsection 3.7.2.~~

Refer to MUAP-10001, "Seismic Design Bases of the US-APWR Standard Plant" (Reference 3H-4) for the model properties for lumped mass stick model of reactor building (R/B) - prestressed concrete containment vessel (PCCV) – Containment Internal structure on a common basemat.

3H.2 Deleted~~Model Properties~~

~~Lumped mass stick models of the R/B, PCCV, and containment internal structure are developed to obtain the seismic response of these structures. The lumped mass stick model for each of these structures is included in a combined model, which represents the R/B-PCCV-containment internal structure resting on their common basemat, as shown in Figure 3H.2-1. Frequency independent SSI lumped parameters representing the stiffness and damping properties of the interaction of the basemat with the underlying subgrade are added at the base of the combined model to simulate a range of different subgrade conditions.~~

~~Figure 3H.2-2 provides elevation views of the R/B, PCCV, and containment internal structure of the overall stick model in the global XZ and YZ planes used for the model. The overall modeling approach and procedures for this lumped mass stick model are discussed in Subsection 3.7.2.3. Node points at the end of rigid outriggers (not shown in Figure 3H.2-1) are also included where appropriate at each floor level in the stick model in order to capture accelerations at the edges/corners of the structures. These response results obtained at the outrigger locations are used to develop ISRS. Comparison of the ISRS obtained from the combined lumped mass stick model versus the detailed finite element (FE) model for static analysis of the R/B-PCCV-containment internal structure are discussed further in Section 3H.3 of this Appendix.~~

~~Table 3H.2-1 provides a physical description of the mass locations for the overall R/B, PCCV, and containment internal structure model, including the node point elevations.~~

~~Mass properties including the associated weight, vertical and horizontal location of the center of gravity, and weight moment of inertia for each node of the PCCV portion of the model are provided in Table 3H.2-2. Table 3H.2-3 presents the PCCV element properties of the stick model elements including the moments of inertia about the global X, Y and Z axes, the shear area, shear area centroid, axial area, and axial area centroid for each stick element. The PCCV shell is a structure with uniform geometry and mass distribution, and the analyzed center of mass at each node point and element corresponds with the center of shear and axial rigidity.~~

~~Mass properties and element properties of the R/B-PCCV-containment internal structure model elements are provided for the containment internal structure portion of the model in Tables 3H.2-4 and 3H.2-5, mass properties and element properties of the model elements are provided for the R/B portion of the model in Tables 3H.2-6 and 3H.2-7, and mass properties and element properties of the model elements are provided for the common basemat portion of the model in Tables 3H.2-8 and 3H.2-9.~~

~~Table 3H.2-10 provides a summation of the weights of the PCCV, R/B, containment internal structure, and basemat portions of the lumped mass stick model.~~

~~The lumped mass stick model of the PS/B as shown in Figure 3H.2-3 is use to obtain the seismic response. Figure 3H.2-4 provides elevation views of the PS/B lumped mass stick model in the NS and EW planes. The overall modeling approach and procedures for this lumped mass stick model are discussed in Subsection 3.7.2.3.~~

~~Mass properties and element properties of the PS/B model elements are provided in Tables 3H.2-11 and 3H.2-12, respectively.~~

~~Table 3H.2-13 presents the properties of the internodal spring elements that are shown in the combined R/B-PCCV-containment internal structure lumped mass stick model in Figure 3H.2-1. The RE42-RE04 and RE41-RE04 springs are modeled as infinitely stiff in order to represent the rigid diaphragm connection between the portions of the R/B roof. The IC07-JC05 spring represents the stiffness of the upper portion of the containment internal structure pressurizer house (located above the operating deck at Elevation 76 ft, 5 in. with respect to the lower portion of the containment internal structure.~~

~~Table 3H.2-14 presents the combined R/B-PCCV-containment internal structure SSI lumped parameter coefficients, soil springs representing the SSI stiffness and damping coefficients representing the dissipation of energy in the SSI system. The SSI lumped parameter coefficients are computed for each of the three flexible generic subgrade conditions using the formulas and general approaches given in American Society of Civil Engineers (ASCE) 4 (Reference 3H-2), discussed in further detail in Subsection 3.7.2.4. The SSI stiffness constants and damping coefficients shown in Table 3H.2-14 are applied at node BB01 representing the bottom center of the basemat, as shown in Figure 3H.2-1. Table 3H.2-15 presents the SSI lumped parameter for the PS/Bs.~~

3H.3 Deleted~~Seismic Analysis Results for Lumped Mass Stick Models~~

~~Table 3H.3-10 presents the results of the PS/B time history analyses by using the methodology described in Subsection 3.7.2.~~

3H.4 References

3H-1 Deleted. ~~Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008.~~

3H-2 Deleted.~~Seismic Analysis of Safety-Related Nuclear Structures. American Society of Civil Engineers, ASCE 4-98, Reston, Virginia, 2000.~~

3H-3 Deleted.~~Combining Responses and Spatial Components in Seismic Response Analysis. Regulatory Guide 1.92, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, July 2006.~~

3H-4 Seismic Design Bases of the US-APWR Standard Plant, MUAP-10001, Revision 2, Mitsubishi Heavy Industries, January 2011.

~~Table 3H.2-1 Description of Mass Locations and Elevations with Respect to Node Points of R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model~~

~~Building~~	~~Mass (Jint)~~	~~Elevation~~	~~Description~~
~~PCCV~~	~~CV11~~	~~EL 230'- 2"~~	~~Wall center at the top of dome~~
	~~CV10~~	~~EL 225'- 0"~~	~~7 feets under the top of dome~~
	~~CV09~~	~~EL 201'- 8"~~	~~Angle of elevation 40 degrees for dome (inside)~~
	~~CV08~~	~~EL 173'- 1"~~	~~Angle of elevation 15 degrees for dome (inside)~~
	~~CV07~~	~~EL 145'- 7"~~	~~The top of polar crane rail~~
	~~CV06~~	~~EL 115'- 6"~~	~~The roof level of MS/FW room~~
	~~CV05~~	~~EL 92'- 2"~~	~~MS penetration~~
	~~CV04~~	~~EL 76'- 5"~~	~~The operation floor level~~
	~~CV03~~	~~EL 68'- 3"~~	~~FW penetration~~
	~~CV02~~	~~EL 50'- 2"~~	~~R/B 3rd floor level~~
	~~CV01~~	~~EL 25'- 3"~~	~~R/B 2nd floor level~~
	~~CV00~~	~~EL 1'-11"~~	~~The upper level of basemat in PCCV~~
~~Containment Internal Structure~~	~~IC09~~	~~EL 139'- 6"~~	~~The upper level of P/R room~~
	~~IC08~~	~~EL 112'- 4"~~	~~The point wall thickness changes in P/R room~~
	~~IC18~~	~~EL 110'- 9"~~	~~P/R support level~~
	~~IC07~~	~~EL 76'- 5"~~	~~The operation floor level in P/R room~~
	~~IC71~~	~~EL 112'- 0"~~	~~The upper level of SG wall~~
	~~IC72~~	~~EL 112'- 0"~~	~~The upper level of SG wall~~
	~~IC61~~	~~EL 96'- 7"~~	~~SG support level~~
	~~IC62~~	~~EL 96'- 7"~~	~~SG support level~~
	~~IC05~~	~~EL 76'- 5"~~	~~The operation floor level~~
	~~IC15~~	~~EL 59'- 2"~~	~~SG support level~~
	~~IC04~~	~~EL 50'- 2"~~	~~R/B 3rd floor level~~
	~~IC14~~	~~EL 45'- 8"~~	~~SG support level~~
	~~IC03~~	~~EL 35'-10.87"~~	~~Reactor vessel support level~~
	~~IC02~~	~~EL 25'- 3"~~	~~R/B 2nd floor level~~
	~~IC01~~	~~EL 16'- 0"~~	~~Pressure header room floor level~~
	~~(IC00)~~	~~EL 1'-11"~~	~~The upper level of basemat in PCCV~~
~~R/B~~	~~FH08~~	~~EL 154'- 6"~~	~~The FH area roof level~~
	~~FH07~~	~~EL 125'- 8"~~	~~The top level of FH area crane rail~~
	~~FH06~~	~~EL 101'- 0"~~	~~The roof level of center building~~
	~~RE05~~	~~EL 115'- 6"~~	~~The roof level of MS/FW room~~
	~~RE04~~	~~EL 101'- 0"~~	~~R/B 5th floor level (MS/FW room)~~
	~~RE41~~	~~EL 101'- 0"~~	~~R/B 5th roof level (west side)~~
	~~RE42~~	~~EL 101'- 0"~~	~~R/B 5th roof level (east side)~~
	~~RE03~~	~~EL 76'- 5"~~	~~R/B 4th floor level (operation floor)~~
	~~RE02~~	~~EL 50'- 2"~~	~~R/B 3rd floor level~~
	~~RE01~~	~~EL 25'- 3"~~	~~R/B 2nd floor level~~
	~~RE00~~	~~EL 3'- 7"~~	~~The upper level of basemat~~
~~Basemat~~	~~BS01~~	~~EL -25'- 0.5"~~	~~The mass level of basemat~~
	~~BB01~~	~~EL -36'- 3"~~	~~The lower level of basemat~~

~~Table 3H.2-2 Mass Properties of Stick Model (PCCV)~~

~~Mass Name~~	~~Elevation (in)~~	~~Weight W ($\times10^6$lb)~~	~~Weight Moment of Inertia ($\times10^{12}$lb·in^2)~~			~~Mass Center (in)~~	
			~~J_{yy} NS~~	~~J_{xx} EW~~	~~J_{zz} Torsional~~	~~X_g NS~~	~~Y_g EW~~
~~CV11~~	~~2,762~~	~~0.887~~	~~0.00749~~	~~0.00749~~	~~0.0146~~	~~0.00~~	~~0.00~~
~~CV10~~	~~2,700~~	~~4.10~~	~~0.415~~	~~0.415~~	~~0.810~~	~~0.00~~	~~0.00~~
~~CV09~~	~~2,420~~	~~7.81~~	~~2.09~~	~~2.09~~	~~4.06~~	~~0.00~~	~~0.00~~
~~CV08~~	~~2,077~~	~~8.49~~	~~3.37~~	~~3.37~~	~~6.59~~	~~0.00~~	~~0.00~~
~~CV07~~	~~1,747~~	~~11.9~~	~~5.12~~	~~5.12~~	~~10.1~~	~~0.00~~	~~0.00~~
~~CV06~~	~~1,386~~	~~9.06~~	~~3.92~~	~~3.92~~	~~7.69~~	~~0.00~~	~~0.00~~
~~CV05~~	~~1,106~~	~~7.53~~	~~3.23~~	~~3.23~~	~~6.39~~	~~0.00~~	~~0.00~~
~~CV04~~	~~917~~	~~4.64~~	~~1.98~~	~~1.98~~	~~3.94~~	~~0.00~~	~~0.00~~
~~CV03~~	~~819~~	~~4.43~~	~~1.89~~	~~1.89~~	~~3.76~~	~~0.00~~	~~0.00~~
~~CV02~~	~~602~~	~~7.27~~	~~3.13~~	~~3.13~~	~~6.17~~	~~0.00~~	~~0.00~~
~~CV01~~	~~303~~	~~8.23~~	~~3.55~~	~~3.55~~	~~6.98~~	~~0.00~~	~~0.00~~

~~Subtotal : 74.36×10^6 (lb)~~

~~Table 3H.2-3 Element Properties of Stick Model (PCCV)~~

~~Element Name~~	~~Torsional Const. I_{zz} ($\times10^{11}$in^4)~~	~~Shear Area ($\times10^5$in^2)~~		~~Moment of Inertia ($\times10^{11}$in^4)~~		~~Shear Center (in)~~		~~Axial Area A_a ($\times10^5$in^2)~~	~~Centroid (in)~~	
		~~A_x NS~~	~~A_y EW~~	~~I_{yy} NS~~	~~I_{xx} EW~~	~~X_s NS~~	~~Y_s EW~~		~~X_c NS~~	~~Y_c EW~~
~~CV11~~	~~0.143~~	~~1.27~~	~~1.27~~	~~0.0715~~	~~0.0715~~	~~0.00~~	~~0.00~~	~~0.0147~~	~~0.0~~	~~0.0~~
~~CV10~~	~~0.837~~	~~1.27~~	~~1.27~~	~~0.418~~	~~0.418~~	~~0.00~~	~~0.00~~	~~0.168~~	~~0.0~~	~~0.0~~
~~CV09~~	~~1.72~~	~~1.27~~	~~1.27~~	~~0.860~~	~~0.860~~	~~0.00~~	~~0.00~~	~~0.814~~	~~0.0~~	~~0.0~~
~~CV08~~	~~2.43~~	~~1.44~~	~~1.44~~	~~1.21~~	~~1.21~~	~~0.00~~	~~0.00~~	~~3.27~~	~~0.0~~	~~0.0~~
~~CV07~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV06~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV05~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV04~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV03~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV02~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~
~~CV01~~	~~2.55~~	~~1.50~~	~~1.50~~	~~1.28~~	~~1.28~~	~~0.00~~	~~0.00~~	~~3.01~~	~~0.0~~	~~0.0~~

~~Table 3H.2-4 Mass Properties of Stick Model (Containment Internal Structure)~~

Mass Name	Elevation (in)	Weight* ($\times10^6$lb)		Weight Moment of Inertia ($\times10^{12}$lb·in^2)			Mass Center (in)	
		W_h	W_v	J_{yy}_NS	J_{xx}_EW	J_{zz} Torsional	X_g_NS	Y_g_EW
IC09	1,674	0.716	0.716	0.00441	0.00797	0.0114	472.5	-0.4
IC08	1,348	2.08	2.08	0.0235	0.0339	0.0331	475.7	0.2
IC18	1,329	0.342	0.011	0.00192	0.00362	0.00543	475.7	0.2
IC07	917	1.07	1.07	0.00746	0.0128	0.0170	435.0	1.4
IC71	1,344	0.817	0.817	0.0276	0.00895	0.0365	97.3	-527.7
IC72	1,344	1.04	1.04	0.0350	0.0114	0.0463	87.5	528.3
IC61	1,159	3.27	2.16	0.112	0.0375	0.146	47.3	-452.8
IC62	1,159	3.30	2.20	0.113	0.0379	0.147	46.6	451.8
IC05	917	17.2	15.1	2.14	2.14	4.23	40.3	8.9
IC15	710	0.441	0.551	0.0511	0.0511	0.102	40.3	8.9
IC04	602	14.9	14.9	1.74	1.74	3.45	23.0	-7.3
IC14	548	2.89	0.220	0.335	0.335	0.670	23.0	-7.3
IC03	430.52	11.7	12.0	1.14	1.14	2.26	-7.2	1.0
IC02	303	17.4	24.3	3.31	3.61	6.60	-7.6	2.1
IC01	192	18.5	18.5	3.51	3.51	7.00	14.6	0.6

~~Sub total : 95.60×10^6 (lb)~~

~~*: W_h is used for horizontal analyses, and W_v is used for vertical analyses~~

~~Table 3H.2-5 Element Properties of Stick Model (Containment Internal Structure)~~

Element Name	Torsional Const. I_{zz} ($\times10^{11}$in^4)	Shear Area ($\times10^5$in^2)		Moment of Inertia ($\times10^{11}$in^4)		Shear Center (in)		Axial Area A_a ($\times10^5$in^2)	Centroid (in)	
		A_x NS	A_y EW	I_{yy} NS	I_{xx} EW	X_s NS	Y_s EW		X_c NS	Y_c EW
C09	0.00694	0.119	0.206	0.00429	0.00662	502.3	0.0	0.411	471.5	0.0
IC08	0.00752	0.140	0.219	0.00525	0.00851	501.0	0.0	0.546	470.9	0.0
IC18	0.00752	0.140	0.219	0.00525	0.00851	501.0	0.0	0.546	470.9	0.0
IC07	-	-	-	-	-	-	-	-	-	-
IC71	0.00263	0.192	0.171	0.00497	0.00250	48.0	-473.3	0.534	39.7	-483.9
IC72	0.00263	0.192	0.171	0.00497	0.00250	48.0	473.3	0.534	39.7	483.9
IC61	0.0347	0.520	0.276	0.0112	0.00565	40.0	-438.2	1.11	39.6	-477.5
IC62	0.0347	0.520	0.276	0.0112	0.00565	40.0	438.2	1.11	39.6	477.5
IC05	0.731	2.19	1.43	0.345	0.271	-15.6	-2.7	4.37	-7.3	-0.5
IC15	0.731	2.19	1.43	0.345	0.271	-15.6	-2.7	4.37	-7.3	-0.5
IC04	0.720	2.11	1.55	0.354	0.286	-17.8	-2.9	4.22	0.5	-5.2
IC14	0.720	2.11	1.55	0.354	0.286	-17.8	-2.9	4.22	0.5	-5.2
IC03	0.646	2.64	2.49	0.412	0.257	-19.1	-0.8	5.52	7.8	-4.7
IC02	1.710	7.57	7.23	0.703	0.367	-15.4	0.0	11.4	-28.5	-2.1
IC01	1.830	12.30	12.00	0.732	0.729	-10.4	0.0	14.9	-44.1	-0.9

~~Table 3H.2-6 Mass Properties of Stick Model (R/B)~~

Mass Name	Elevation (in)	Weight W ($\times10^6$ lb)	Weight Moment of Inertia ($\times10^{12}$ lb·in^2)			Mass Center (in)	
			J_{yy} NS	J_{xx} EW	J_{zz} Torsional	X_g NS	Y_g EW
FH08	1,854	6.25	0.303	2.38	2.68	-1,400.0	209.3
FH07	1,508	4.52	0.218	1.72	1.94	-1,445.9	224.5
FH06	1,212	4.68	0.232	1.78	2.01	-1,449.7	208.9
RE41	1,212	9.45	6.51	0.20	6.7	-3,41.5	-924.8
RE42	1,212	7.59	2.99	0.174	3.16	41.8	948
RE05	1,386	19.2	1.34	10.5	11.8	1,385.0	74.0
RE04	1,212	16.6	1.15	9.06	10.2	1,434.5	-94.0
RE03	917	67.7	77.6	37.1	115	119.4	24.1
RE02	602	72.4	83.1	39.7	123	-66.0	-35.1
RE01	303	68.6	78.7	37.6	116	-59.7	-43.8

~~Subtotal: 277.0×10^6 (lb)~~

~~Table 3H.2-7 Element Properties of Stick Model (R/B)~~

Element Name	Torsional Const. I_{zz} ($\times10^{11}$ in^4)	Shear Area ($\times10^5$ in^2)		Moment of Inertia ($\times10^{11}$ in^4)		Shear Center (in)		Axial Area A_a ($\times10^5$ in^2)	Centroid (in)	
		A_x NS	A_y EW	I_{yy} NS	I_{xx} EW	X_s NS	Y_s EW		X_c NS	Y_c EW
FH08	0.488	0.365	0.898	0.0176	0.342	-1,437.5	210.0	1.51	-1,440.9	220.4
FH07	0.757	0.608	0.898	0.0293	0.342	-1,437.5	210.0	1.8	-1,441.6	220.0
FH06	0.761	0.616	0.851	0.0308	0.279	-1,454.5	222.8	1.75	-1,448.8	169.6
RE41	0.610	1.19	0.479	0.586	0.00795	-424.2	-1,191.0	1.98	-342.9	-1,112.7
RE42	0.103	0.816	0.533	0.157	0.00957	637.3	1,165.6	1.53	403.1	1,038.5
RE05	1.86	2.04	1.6	0.182	0.596	1,686.1	121.2	3.84	1,533.6	82.1
RE04	2.08	2.13	2.17	0.185	0.625	1,542.2	87.1	4.41	1,499.7	51.4
RE03	21.5	7.02	7.39	4.44	3.08	-283.4	-40.2	13.7	-68.8	-12.1
RE02	22.7	7.95	7.79	4.8	2.8	-159.7	-72.9	15	-64.1	-49.1
RE01	23.6	8.09	8.29	4.6	2.93	-69.0	-34.7	15.6	-0.4	-25.1

~~Table 3H.2-8 Mass Properties of Stick Model (Basemat)~~

Mass Name	Elevation (in)	Weight W ($\times10^6$lb)	Weight Moment of Inertia ($\times10^{12}$lb·in2)			Mass Center (in)	
			Jyy NS	Jxx EW	Jzz Torsional	Xg NS	Yg EW
CV00	23	3.94	1.68	1.68	3.34	0.0	0.0
RE00	43	117	134	64.3	198	14.9	-3.0
IC00	23	21.8	4.16	4.16	8.28	14.6	0.6
BS01	-313.0	187	216	104	317	29.1	9.0
BB01*	-435	-	-	-	-	0.0	0.0

~~Subtotal: 329.7×10^6 (lb)~~

~~*: BB01 is a subordinate point of BS01.~~

~~Table 3H.2-9 Element Properties of Stick Model (Basemat)~~

Element Name	Torsional Const. Izz ($\times10^{11}$in4)	Shear Area ($\times10^5$in2)		Moment of Inertia ($\times10^{11}$in4)		Shear Center (in)		Axial Area Aa ($\times10^5$in2)	Centroid (in)	
		Ax NS	Ay EW	Iyy NS	Ixx EW	Xs NS	Ys EW		Xc NS	Yc EW
RE00	41.8	25.6	22.7	15.2	12.8	34.92	-36.8	30.7	27.3	-27.2

~~Note: CV00, RE00, and IC00 are linked by rigid beam elements.~~

~~Table 3H.2-10 Summation of Lumped Mass Stick Model Weights~~

Mass Name	Weight (×106lb) W
PCCV	74.36
R/B	277.0
Containment Internal Structure	95.60
Basemat	329.7
Total	776.7

~~Table 3H.2-11 Mass Properties of Stick Model (PS/B)~~

~~Mass Name~~	~~Elevation (in)~~	~~Weight (×106lb) W~~	~~Weight Moment of Inertia (×1012lb·in2)~~			~~Mass Center (in)~~	
			~~Jyy NS~~	~~Jxx EW~~	~~Jzz Torsional~~	~~xg NS~~	~~yg EW~~
~~PSB2~~	~~474.0~~	~~8.74~~	~~0.508~~	~~1.39~~	~~1.89~~	~~394.6~~	~~656.7~~
~~PSB1~~	~~43.0~~	~~11.4~~	~~0.667~~	~~1.81~~	~~2.46~~	~~398.2~~	~~649.9~~
~~BSTP*~~	~~-316.0~~	~~—~~	~~—~~	~~—~~	~~—~~	~~—~~	~~—~~
~~BASE~~	~~-375.5~~	~~17.0~~	~~1.02~~	~~2.74~~	~~3.68~~	~~399.5~~	~~648.0~~
~~BSBM*~~	~~-435.0~~	~~—~~	~~—~~	~~—~~	~~—~~	~~—~~	~~—~~

~~Subtotal : 37.16×10^6 (lb)~~

~~Note * = BSTP and BSBM are subordinate points of BASE.~~

~~Table 3H.2-12 Element Properties of Stick Model (PS/B)~~

~~Element Name~~	~~Torsional Const. ($\times10^{11}$in4) Izz~~	~~Shear Area ($\times10^{5}$in2)~~		~~Moment of Inertia ($\times10^{11}$in4)~~		~~Shear Center (in)~~		~~Axial Area ($\times10^{5}$in2) Aa~~	~~Centroid (in)~~	
		~~Ax NS~~	~~Ay EW~~	~~Iyy NS~~	~~Ixx EW~~	~~Xs NS~~	~~Ys EW~~		~~xc NS~~	~~yc EW~~
~~PSB2~~	~~0.297~~	~~0.555~~	~~0.577~~	~~0.0619~~	~~0.207~~	~~397.0~~	~~689.6~~	~~1.13~~	~~417.8~~	~~703.9~~
~~PSB1~~	~~0.412~~	~~0.962~~	~~0.936~~	~~0.0922~~	~~0.273~~	~~393.2~~	~~664.5~~	~~1.83~~	~~407.9~~	~~652.2~~

~~Table 3H.2-13 Spring Properties for Internodal Spring Elements~~

	~~Location~~	~~Sort~~		~~Spring Value~~	~~Damping Value~~
~~Containment Internal Structure~~	~~Area at lower pressurizer support IC07-JC05~~ [(2)]	~~NS~~	~~Horizontal~~	~~infinity~~	~~h=5%~~
			~~Rotational~~	~~6.75×10^{12} lb·in/rad~~	
		~~EW~~	~~Horizontal~~	~~infinity~~	
			~~Rotational~~	~~1.09×10^{13} lb·in/rad~~	
		~~Vertical~~		~~infinity~~	
		~~Torsion~~		~~infinity~~	
~~R/B~~	~~Roof area 1~~ [(1)] ~~(RE42-RE04)~~	~~NS~~	~~Horizontal~~	~~infinity~~	~~h=7%~~
		~~EW~~	~~Horizontal~~	~~infinity~~	
	~~Roof area 2~~ [(1)] ~~(RE41-RE04)~~	~~NS~~	~~Horizontal~~	~~infinity~~	
		~~EW~~	~~Horizontal~~	~~infinity~~	

~~Notes:~~

1. ~~RE41, RE42, and RE04 are linked by rigid translational springs. No link elements are set between RE41, RE 42 and FH06. See Figure 3H.2-1 for the overall configuration of the R/B-PCCV-Containment Internal Structure lumped mass stick model.~~

2. ~~JC05 is a subordinate point of IC05 and located at the same coordinate as IC07. JC05 and IC07 are connected by internodal spring elements shown in the above table.~~

~~Table 3H.2-14 SSI Lumped Parameters~~
~~(R/B, PCCV, and Containment Internal Structure)~~

~~(Horizontal)~~

~~Direction~~		~~Horizontal~~		~~Rotational~~	
		~~Spring Constant ($\times10^8$ lb/in)~~	~~Damping Coefficient ($\times10^7$ lb-s/in)~~	~~Spring Constant ($\times10^{14}$ lb-in/rad)~~	~~Damping Coefficient ($\times10^{13}$ lb-in-s/rad)~~
~~NS~~ ~~K_x: Horizontal~~ ~~K_{yy}: Rotational~~	~~Soft~~	~~1.89~~	~~0.948~~	~~7.83~~	~~3.39~~
	~~Medium 1~~	~~26.4~~	~~3.78~~	~~105.~~	~~13.1~~
	~~Medium 2~~	~~98.2~~	~~7.56~~	~~389.~~	~~26.3~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~			
~~EW~~ ~~K_y: Horizontal~~ ~~K_{xx}: Rotational~~	~~Soft~~	~~2.05~~	~~1.02~~	~~4.57~~	~~1.74~~
	~~Medium 1~~	~~28.6~~	~~4.09~~	~~61.0~~	~~6.65~~
	~~Medium 2~~	~~106.~~	~~8.16~~	~~227.~~	~~13.4~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~			

~~(Vertical)~~

~~Direction~~		~~Spring Constant ($\times10^8$ lb/in)~~	~~Damping Coefficient ($\times10^7$ lb-s/in)~~
~~UD~~ ~~K_z~~	~~Soft~~	~~2.62~~	~~3.23~~
	~~Medium 1~~	~~35.0~~	~~12.3~~
	~~Medium 2~~	~~130.~~	~~24.6~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~	

~~(Torsional)~~

~~Direction~~		~~Spring Constant ($\times10^{14}$ lb-in/rad)~~	~~Damping Coefficient ($\times10^{13}$ lb-in-s/rad)~~
~~Torsional~~ ~~K_{zz}~~	~~Soft~~	~~7.24~~	~~1.64~~
	~~Medium 1~~	~~105.~~	~~6.82~~
	~~Medium 2~~	~~389.~~	~~13.7~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~	

~~* The points located at the upper level of the basemat (RE00, IC00, CV00) are considered as the fixed end points when a fixed base assumption is adopted.~~

~~Table 3H.2-15 SSI Lumped Parameters (PS/B)~~

~~(Horizontal)~~

~~Direction~~		~~Horizontal~~		~~Rotational~~	
		~~Spring Constant~~ ~~($\times 10^8$ lb/in)~~	~~Damping Coefficient~~ ~~($\times 10^7$ lb s/in)~~	~~Spring Constant~~ ~~($\times 10^{14}$ lb in/rad)~~	~~Damping Coefficient~~ ~~($\times 10^{13}$ lb in s/rad)~~
~~NS~~ ~~K_x:Horizontal~~ ~~K_{yy}Rotational~~	~~Soft~~	~~0.712~~	~~0.120~~	~~0.162~~	~~0.0208~~
	~~Medium 1~~	~~9.94~~	~~0.480~~	~~2.17~~	~~0.0797~~
	~~Medium 2~~	~~36.9~~	~~0.960~~	~~8.06~~	~~0.160~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~			
~~EW~~ ~~K_y:Horizontal~~ ~~K_{xx}:Rotational~~	~~Soft~~	~~0.655~~	~~0.108~~	~~0.356~~	~~0.0542~~
	~~Medium 1~~	~~9.14~~	~~0.450~~	~~4.76~~	~~0.209~~
	~~Medium 2~~	~~34.0~~	~~0.900~~	~~17.7~~	~~0.421~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~			

~~(Vertical)~~

~~Direction~~		~~Spring Constant~~ ~~($\times 10^8$ lb/in)~~	~~Damping Coefficient~~ ~~($\times 10^7$ lb s/in)~~
~~UD~~ ~~K_z~~	~~Soft~~	~~0.928~~	~~0.397~~
	~~Medium 1~~	~~12.4~~	~~1.52~~
	~~Medium 2~~	~~46.1~~	~~3.03~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~	

~~(Torsional)~~

~~Direction~~		~~Spring Constant~~ ~~($\times 10^{14}$ lb in/rad)~~	~~Damping Coefficient~~ ~~($\times 10^{13}$ lb in s/rad)~~
~~Torsional~~ ~~K_{zz}~~	~~Soft~~	~~0.317~~	~~0.0251~~
	~~Medium 1~~	~~4.59~~	~~0.105~~
	~~Medium 2~~	~~17.0~~	~~0.210~~
	~~Hard Rock~~	~~Fixed Base Assumption*~~	

~~* The points located at the top of the basemat (BSTP) is considered as the fixed end point when a fixed base assumption is adopted.~~

~~Table 3H.3-1 Deleted~~

~~Table 3H.3-2 Deleted~~

~~Table 3H.3-3 Deleted~~

~~Table 3H.3-4 Deleted~~

~~Table 3H.3-5 Deleted~~

~~Table 3H.3-6 Deleted~~

~~Table 3H.3-7 Deleted~~

~~**Table 3H.3-8 Deleted**~~

~~Table 3H.3-9 Deleted~~

~~Table 3H.3-10 Lumped Mass Stick Model Design Shear Forces and Moments~~

~~Model~~	~~Mass Node~~	~~Elevation (in)~~	~~N-S Direction~~		~~E-W Direction~~	
			~~Shear Force (kip)~~	~~Moment (kip-ft)~~	~~Shear Force (kip)~~	~~Moment (kip-ft)~~
~~PS/B~~	~~PSB2~~	~~474~~	~~7,170~~	~~285,000~~	~~7,450~~	~~326,000~~
	~~PSB1~~	~~43~~	~~12,500~~	~~685,000~~	~~13,100~~	~~744,000~~

~~Notes:~~

1) ~~The forces and moments shown above envelope all four generic subgrade conditions and are applied to the FE models for structural design as described in Section 3.8.~~

2) ~~The forces and moments are obtained by combination of the three orthogonal directions used in the model by SRSS or the Newmark 100% - 40% - 40%~~

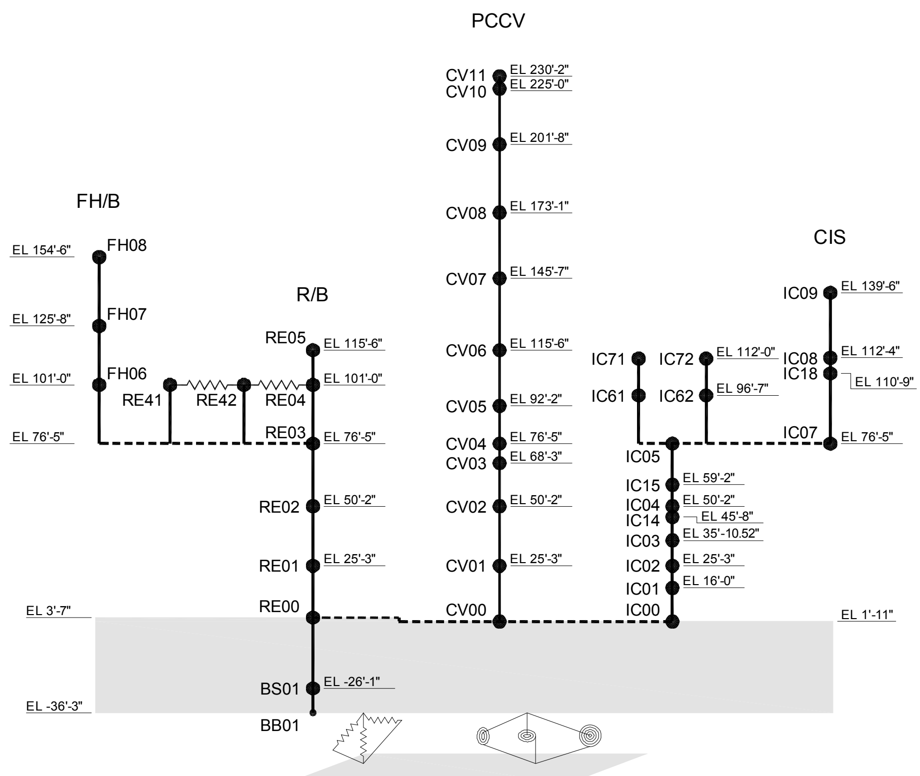
~~Table 3H.3-11 Deleted~~

~~Table 3H.3-12 Deleted~~

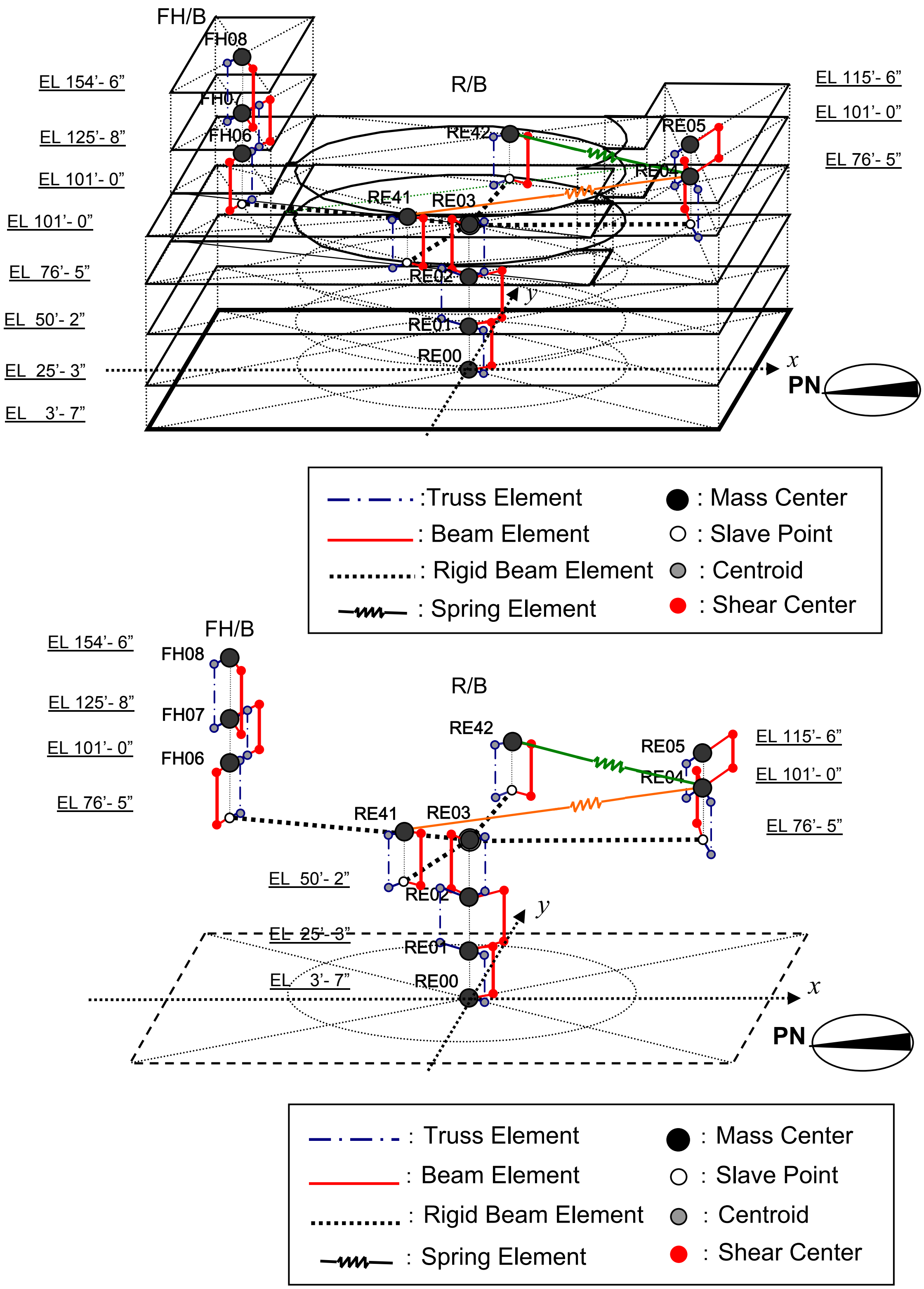
~~Table 3H.3-13 Deleted~~

~~Table 3H.3-14 Deleted~~

~~Security-Related Information – Withheld Under 10 CFR 2.390~~



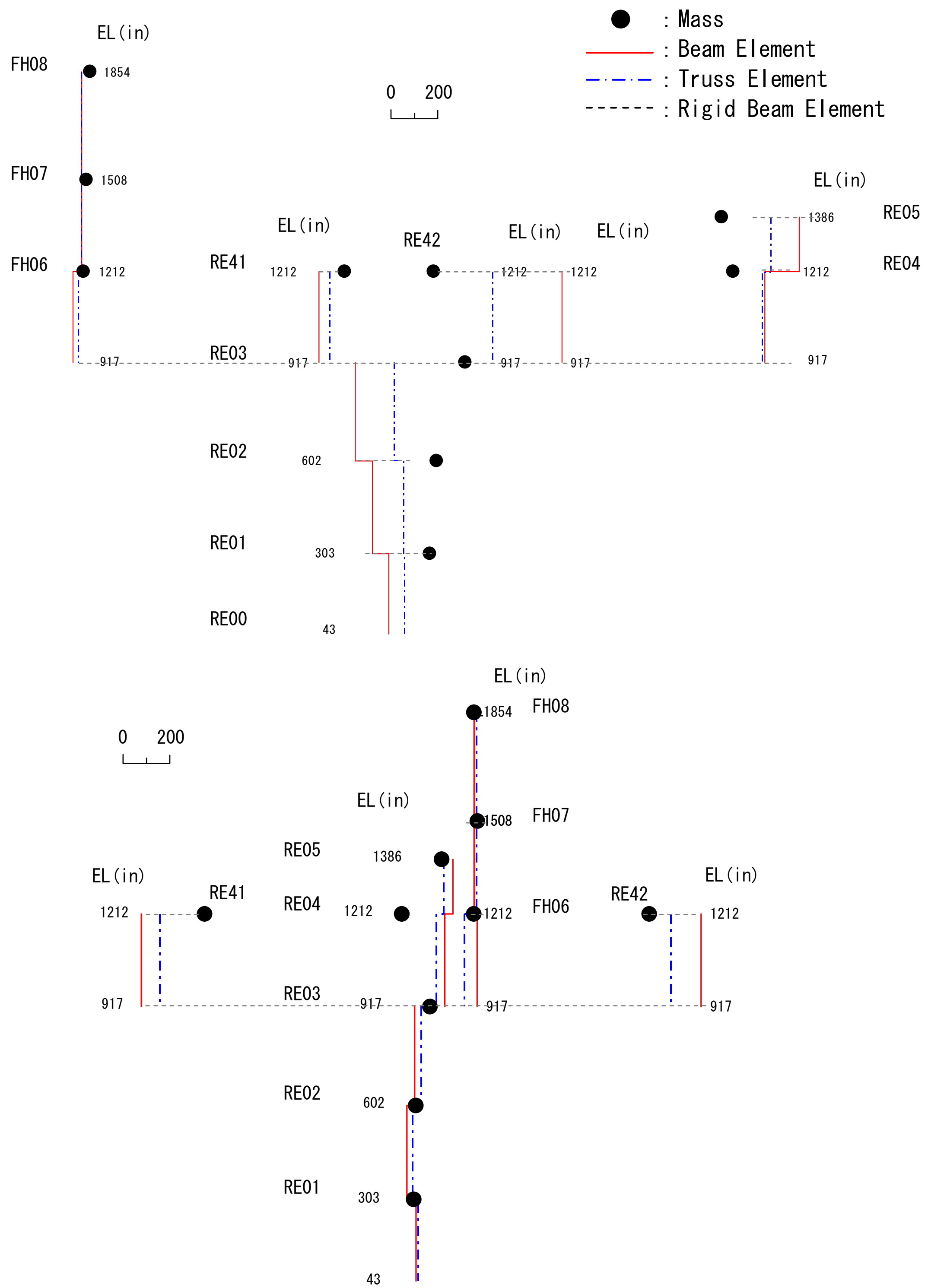
~~**Figure 3H.2-1 Lumped Mass Stick Model for R/B-PCCV-Containment Internal Structure (Sheet 1 of 3)**~~



~~Figure 3H.2-1 Lumped Mass Stick Model for R/B-PCCV-Containment Internal Structure~~
~~(Sheet 2 of 3)~~

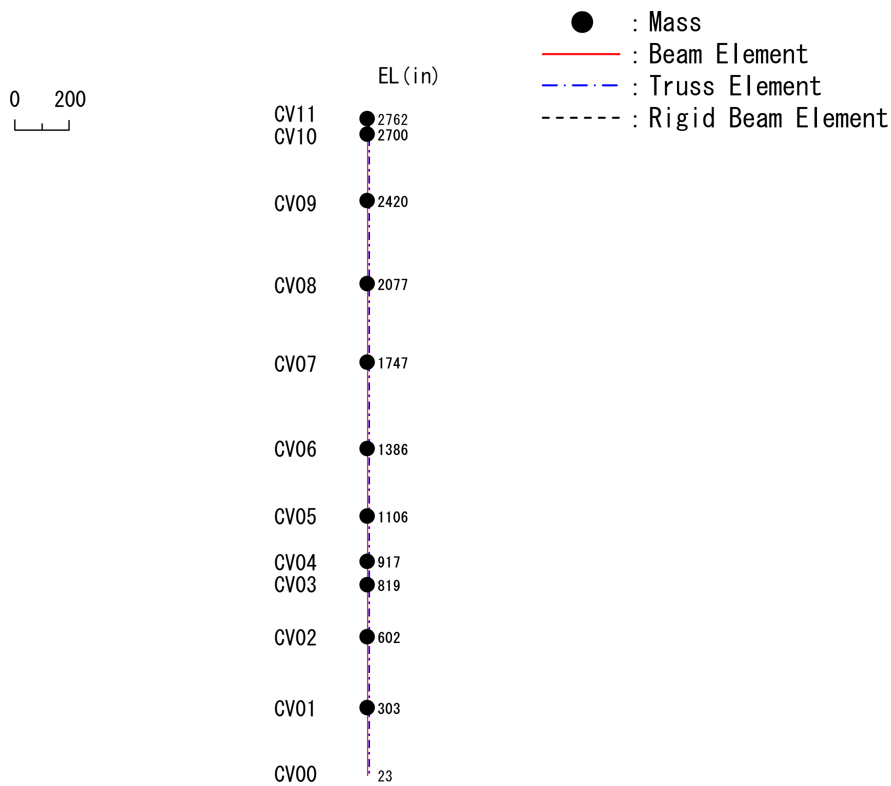
~~Security-Related Information – Withheld Under 10 CFR 2.390~~

~~**Figure 3H.2-1 Lumped Mass Stick Model for R/B-PCCV-Containment Internal Structure (Sheet 3 of 3)**~~



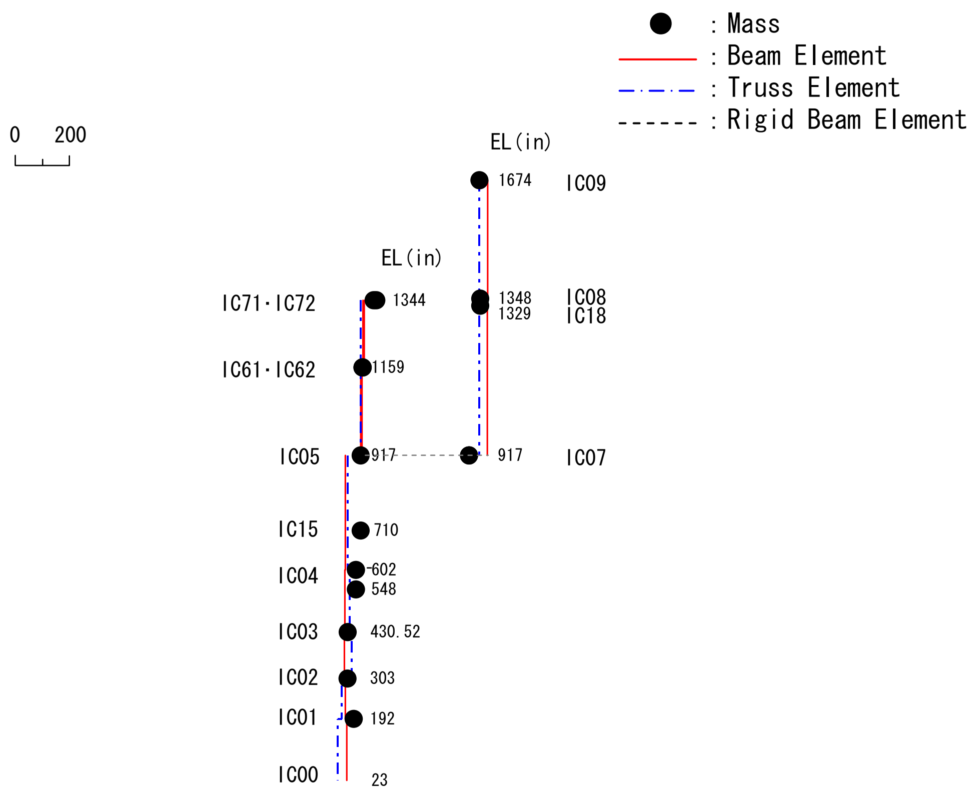
~~Note: This upper view is a north-south elevation view of the R/B model looking east (XZ plane of global model) with node point elevations given in inches. The lower view is an east-west elevation of the R/B model looking north (YZ plane of global model) with node point elevations given in inches.~~

~~**Figure 3H.2-2 Elevation Views of Lumped Mass Stick Model (Sheet 1 of 4)**~~



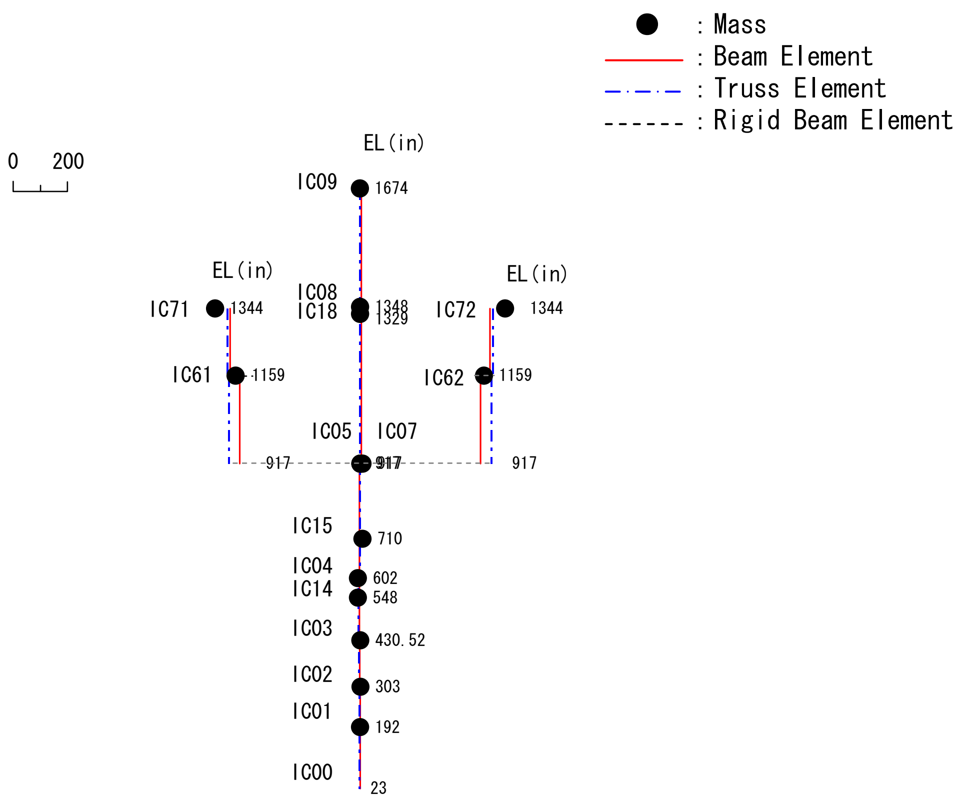
~~Note: This view is an elevation view of the PCCV model with node point elevations given in inches~~

~~**Figure 3H.2-2 Elevation Views of Lumped Mass Stick Model (Sheet 2 of 4)**~~



~~Note: This view is a north-south elevation view looking east at the containment internal structure model with node point elevations given in inches.~~

~~**Figure 3H.2-2 Elevation Views of Lumped Mass Stick Model (Sheet 3 of 4)**~~



~~Note: This view is an east-west elevation view of the containment internal structure model with node point elevations given in inches.~~

~~**Figure 3H.2-2 Elevation Views of Lumped Mass Stick Model (Sheet 4 of 4)**~~

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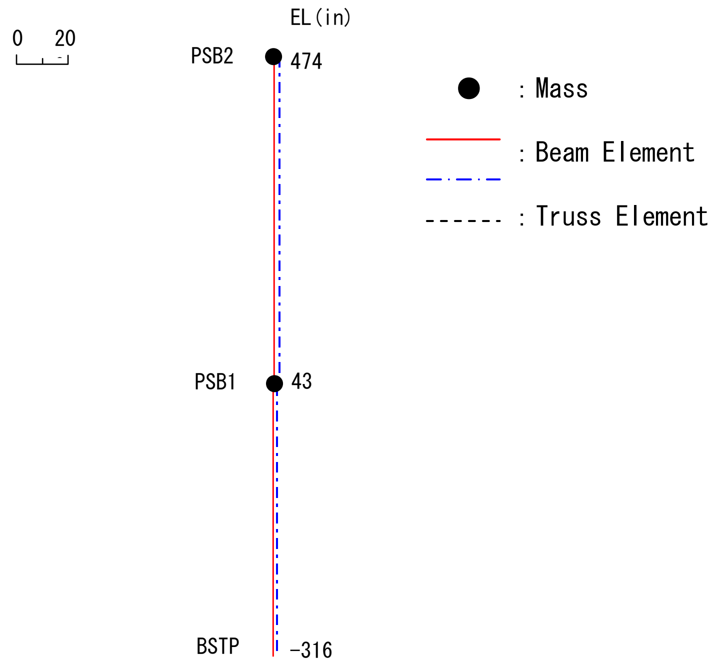
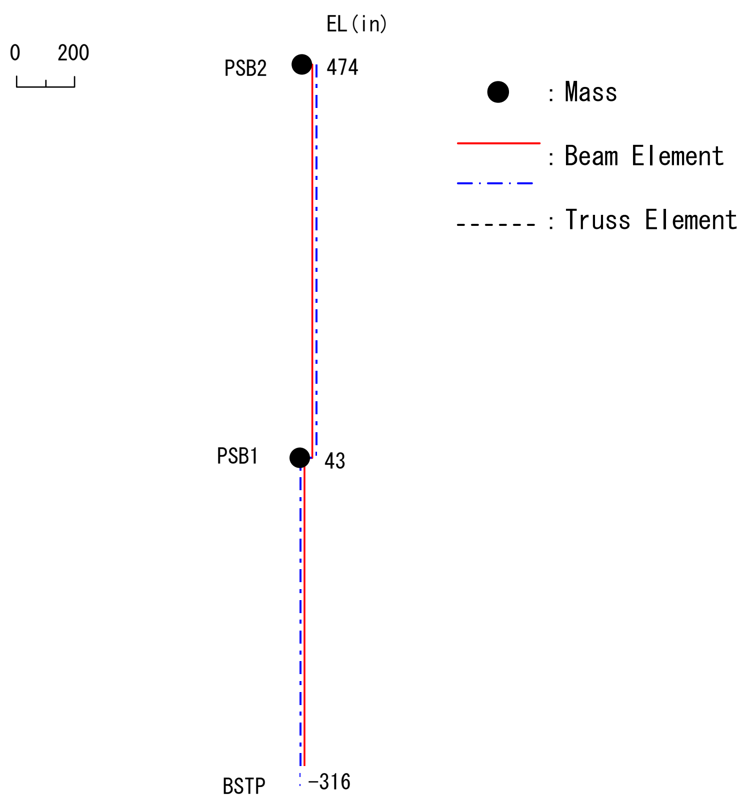
~~**Figure 3H.2-3 Lumped Mass Stick Model for PS/B**~~



~~(NS Direction)~~



~~(EW Direction)~~

~~Figure 3H.2-4 Elevation Views of PS/Bs Lumped Mass Stick Model~~

~~Figure 3H.3-1 Deleted~~

~~Figure 3H.3-2 Deleted~~

~~Figure 3H.3-3 Deleted~~

~~Figure 3H.3-4 Deleted~~

3I. In-Structure Response Spectra

3I.1 Introduction

Refer to ~~MUAP-08005, "Dynamic Analysis of the Coupled RCL-R/B-PCCV-CIS Lumped Mass Stick Model" (Reference 3I-1), and MUAP-08002, "Enhanced Information for PS/B Design" (Reference 3I-2),~~ MUAP-10006, "Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant" (Reference 3I-3) for the in-structure response spectra (ISRS) for various buildings and elevations of the US-APWR standard plant.

3I.2 References

3I-1 ~~Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008.~~ Deleted.

3I-2 ~~Enhanced Information for PS/B Design, MUAP-08002, Mitsubishi Heavy Industries, Ltd., February 2008.~~ Deleted.

3I-3 Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant, MUAP-10006, Rev. 1, January 2011.

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Figure 3J-1 R/B Structural Drawings
(Sheet 3 of 14)

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Figure 3J-1 R/B Structural Drawings
(Sheet 5 of 14)

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3J-1 R/B Structural Drawings
(Sheet 9 of 14)

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Figure 3J-1 R/B Structural Drawings
(Sheet 13 of 14)

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 1 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
1	RCS-AOV-132	Air Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
2	RCS-AOV-148	Air Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
3	RCS-AOV-138	Air Operated Valve	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
4	CVS-MOV-151	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
5	CVS-MOV-152	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
6	CVS-MOV-204	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
7	CVS-MOV-178A	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
8	CVS-MOV-178B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
9	CVS-AOV-006	Air Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
10	CVS-MOV-178C	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
11	CVS-MOV-178D	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-24~~127-08	above flood elevation	0.69	
12	SIS-MPP-001A	A-Safety Injection Pump	R/B RCA	E	-26'-4"	FA2-113-01	N/A	-	1
13	SIS-MPP-001B	B-Safety Injection Pump	R/B RCA	E	-26'-4"	FA2-114-01	N/A	-	1
14	SIS-MPP-001C	C-Safety Injection Pump	R/B RCA	W	-26'-4"	FA2-115-01	N/A	-	1
15	SIS-MPP-001D	D-Safety Injection Pump	R/B RCA	W	-26'-4"	FA2-116-01	N/A	-	1
16	SIS-MOV-001A	Motor Operated Valve	R/B RCA	E	-8'-7"	FA2-~~120-02~~154-01	N/A	-	1
17	SIS-MOV-001B	Motor Operated Valve	R/B RCA	E	-8'-7"	FA2-151-01	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 2 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
18	SIS-MOV-009A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
19	SIS-MOV-009B	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
20	SIS-MOV-001C	Motor Operated Valve	R/B RCA	W	-8'-7"	FA2-152-01	N/A	-	1
21	SIS-MOV-001D	Motor Operated Valve	R/B RCA	W	-8'-7"	FA2-153-01	N/A	-	1
22	SIS-MOV-009C	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
23	SIS-MOV-009D	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
24	SIS-AOV-114	Air Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
25	RHS-MPP-001A	A-Containment Spray/Residual Heat Removal Pump	R/B RCA	E	-26'-4"	FA2-113-02	N/A	-	1
26	RHS-MPP-001B	B-Containment Spray/Residual Heat Removal Pump	R/B RCA	E	-26'-4"	FA2-114-02	N/A	-	1
27	RHS-MPP-001C	C-Containment Spray/Residual Heat Removal Pump	R/B RCA	W	-26'-4"	FA2-115-02	N/A	-	1
28	RHS-MPP-001D	D-Containment Spray/Residual Heat Removal Pump	R/B RCA	W	-26'-4"	FA2-116-02	N/A	-	1
29	RHS-MHX-001A	A-Containment Spray/Residual Heat Removal Heat Exchanger	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
30	RHS-MHX-001B	B-Containment Spray/Residual Heat Removal Heat Exchanger	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1
31	RHS-MHX-001C	C-Containment Spray/Residual Heat Removal Heat Exchanger	R/B RCA	W	3'-7"	FA2-152-03	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 3 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
32	RHS-MHX-001D	D-Containment Spray/Residual Heat Removal Heat Exchanger	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
33	RHS-MOV-021A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-~~120-06~~154-15	above flood elevation	0.53	
34	RHS-MOV-021B	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
35	RHS-HCV-023	Hand Control Valve	R/B RCA	E	3'-7"	FA2-151-~~01~~03	N/A	-	1
36	RHS-FCV-021	Flow Control Valve	R/B RCA	E	3'-7"	FA2-151-01	N/A	-	1
37	RHS-MOV-021C	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
38	RHS-MOV-021D	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
39	RHS-HCV-033	Hand Control Valve	R/B RCA	W	3'-7"	FA2-152-~~01~~03	N/A	-	1
40	RHS-FCV-031	Flow Control Valve	R/B RCA	W	3'-7"	FA2-152-01	N/A	-	1
41	CSS-MOV-004A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
42	CSS-MOV-004B	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
43	CSS-MOV-001A	Motor Operated Valve	R/B RCA	E	-8'-7	FA2-~~120-02~~154-01	N/A	-	1
44	CSS-MOV-001B	Motor Operated Valve	R/B RCA	E	-8'-7	FA2-151-01	N/A	-	1
45	CSS-MOV-004C	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
46	CSS-MOV-004D	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
47	CSS-MOV-001C	Motor Operated Valve	R/B RCA	W	-8'-7	FA2-152-01	N/A	-	1
48	CSS-MOV-001D	Motor Operated Valve	R/B RCA	W	-8'-7	FA2-153-01	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding
(Sheet 4 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
49	CSS-MOV-011	Motor Operated Valve	R/B RCA	E	3'-7"	FA2-151-~~01~~04	~~N/A~~above flood elevation	0.69	1
50	NCS-MOV-145A	Motor Operated Valve	R/B RCA	E	3'-7"	FA2-~~117-08~~209-03	above flood elevation	0.69	
51	NCS-MOV-438A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
52	NCS-MOV-145B	Motor Operated Valve	R/B RCA	E	3'-7"	FA2-151-04	above flood elevation	0.69	
53	NCS-MOV-145C	Motor Operated Valve	R/B RCA	W	3'-7"	FA2-152-04	above flood elevation	0.88	
54	NCS-MOV-145D	Motor Operated Valve	R/B RCA	W	3'-7"	FA2-~~117-07~~128-02	above flood elevation	0.88	
55	NCS-MOV-232A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
56	NCS-MOV-232B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
57	NCS-MOV-233A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.54	
58	NCS-MOV-233B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
59	NCS-MOV-234A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
60	NCS-MOV-234B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
61	NCS-MOV-511	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
62	NCS-MOV-517	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
63	NCS-MOV-402A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
64	NCS-MOV-531	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
65	NCS-MOV-537	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 5 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
68	NCS-MOV-445B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
69	NCS-MOV-448A	Motor Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
70	NCS-MOV-448B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
71	NCS-MOV-438B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
72	LMS-AOV-053	Air Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
73	LMS-AOV-056	Air Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
74	LMS-AOV-060	Air Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
75	LMS-LCV-010B	Level Control Valve	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
76	LMS-AOV-105	Air Operated Valve	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
77	PSS-MOV-031A	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
78	PSS-MOV-031B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
79	PSS-MOV-052A	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-23~~322-01	above flood elevation	0.69	
80	PSS-MOV-052B	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-~~117-23~~322-01	above flood elevation	0.69	
81	PSS-MOV-052C	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-322-01	above flood elevation	0.69	
82	PSS-MOV-052D	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-322-01	above flood elevation	0.69	
83~~1~~	PSS-AOV-063	Air Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
84~~2~~	PSS-MOV-071	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 6 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
85~~3~~	SGS-AOV-031A	Air Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
86~~4~~	SGS-AOV-031B	Air Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
87~~5~~	SGS-AOV-031C	Air Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
88~~6~~	SGS-AOV-031D	Air Operated Valve	R/B RCA	E	25'-3"	FA2-151-06	above flood elevation	0.53	
89~~7~~	RWS-MOV-004	Motor Operated Valve	R/B RCA	E	3'-7"	FA2-~~117-09~~211-01	above flood elevation	0.69	
90~~88~~	RWS-AOV-022	Air Operated Valve	R/B RCA	E	3'-7"	FA2-~~117-09~~211-01	above flood elevation	0.69	
91~~89~~	IAS-MOV-002	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-152-06	above flood elevation	0.69	
92~~0~~	RMS-MOV-002	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
93~~1~~	RMS-MOV-003	Motor Operated Valve	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
94~~2~~	VRS-MFU-001A	A-Annulus Emergency Exhaust Filtration Unit	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
95~~3~~	VRS-MFU-001B	B-Annulus Emergency Exhaust Filtration Unit	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
96~~4~~	VRS-MFN-001A	A-Annulus Emergency Exhaust Filtration Unit Fan	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
97~~5~~	VRS-MFN-001B	B-Annulus Emergency Exhaust Filtration Unit Fan	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
98~~6~~	VRS-EHD-001A	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
99~~7~~	VRS-EHD-001B	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
100~~98~~	VRS-EHD-002A	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 7 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
101~~99~~	VRS-EHD-002B	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
102~~0~~	VRS-EHD-003A	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
103~~1~~	VRS-EHD-003B	Electro Hydraulic~~Motor~~ Operated Damper	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
104~~2~~	VRS-MAH-301A	A-Safeguard Component Area Air Handling Unit	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
105~~3~~	VRS-MAH-301B	B-Safeguard Component Area Air Handling Unit	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
106~~4~~	VRS-MAH-301C	C-Safeguard Component Area Air Handling Unit	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
107~~5~~	VRS-MAH-301D	D-Safeguard Component Area Air Handling Unit	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
108~~6~~	VRS-MFN-301A	A-Safeguard Component Area Air Handling Unit Fan	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
109~~7~~	VRS-MFN-301B	B-Safeguard Component Area Air Handling Unit Fan	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
110~~08~~	VRS-MFN-301C	C-Safeguard Component Area Air Handling Unit Fan	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
111~~09~~	VRS-MFN-301D	D-Safeguard Component Area Air Handling Unit Fan	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
112~~0~~	VRS-MCL-301A	A-Safeguard Component Area Air Handling Unit Cooling Coil	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
113~~1~~	VRS-MCL-301B	B-Safeguard Component Area Air Handling Unit Cooling Coil	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 8 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
114~~2~~	VRS-MCL-301C	C-Safeguard Component Area Air Handling Unit Cooling Coil	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
115~~3~~	VRS-MCL-301D	D-Safeguard Component Area Air Handling Unit Cooling Coil	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
116~~4~~	VRS-MEH-301A	A-Safeguard Component Area Air Handling Unit Electric Heating Coil	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
117~~5~~	VRS-MEH-301B	B-Safeguard Component Area Air Handling Unit Electric Heating Coil	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
118~~6~~	VRS-MEH-301C	C-Safeguard Component Area Air Handling Unit Electric Heating Coil	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
119~~7~~	VRS-MEH-301D	D-Safeguard Component Area Air Handling Unit Electric Heating Coil	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
120~~18~~	VRS-MOD-301A	Motor Operated Damper	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
121~~19~~	VRS-MOD-301B	Motor Operated Damper	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
122~~0~~	VRS-MOD-301C	Motor Operated Damper	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
123~~1~~	VRS-MOD-301D	Motor Operated Damper	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
124~~2~~	VRS-MOD-302A	Motor Operated Damper	R/B RCA	E	3'-7"	FA2-~~120-05~~154-04	N/A	-	1
125~~3~~	VRS-MOD-302B	Motor Operated Damper	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
126~~4~~	VRS-MOD-302C	Motor Operated Damper	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
127~~5~~	VRS-MOD-302D	Motor Operated Damper	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 9 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
128~~6~~	VRS-MAH-541A	A-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
129~~7~~	VRS-MAH-541B	B-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
130~~28~~	VRS-MFN-541A	A-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Fan	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
131~~29~~	VRS-MFN-541B	B-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Fan	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
132~~0~~	VRS-MCL-541A	A-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Cooling Coil	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
133~~1~~	VRS-MCL-541B	A-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Cooling Coil	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
134~~2~~	VRS-MCL-541C	B-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Cooling Coil	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
135~~3~~	VRS-MCL-541D	B-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Cooling Coil	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 10 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
136~~4~~	VRS-MEH-541A	A-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Electric Heating Coil	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
137~~5~~	VRS-MEH-541B	B-Annulus Emergency Exhaust Filtration Unit Area Air Handling Unit Electric Heating Coil	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
138~~6~~	VRS-MAH-551A	A-Penetration Area Air Handling Unit	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
139~~7~~	VRS-MAH-551B	B-Penetration Area Air Handling Unit	R/B RCA	E	50'-2"	FA2-409-01	above flood elevation	0.58	
140~~38~~	VRS-MAH-551C	C-Penetration Area Air Handling Unit	R/B RCA	W	50'-2"	FA2-410-01	above flood elevation	0.76	
141~~39~~	VRS-MAH-551D	D-Penetration Area Air Handling Unit	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
142~~0~~	VRS-MFN-551A	A-Penetration Area Air Handling Unit Fan	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
143~~1~~	VRS-MFN-551B	B-Penetration Area Air Handling Unit Fan	R/B RCA	E	50'-2"	FA2-409-01	above flood elevation	0.58	
144~~2~~	VRS-MFN-551C	C-Penetration Area Air Handling Unit Fan	R/B RCA	W	50'-2"	FA2-410-01	above flood elevation	0.76	
145~~3~~	VRS-MFN-551D	D-Penetration Area Air Handling Unit Fan	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
146~~4~~	VRS-MCL-551A	A-Penetration Area Air Handling Unit Cooling Coil	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
147~~5~~	VRS-MCL-551B	B-Penetration Area Air Handling Unit Cooling Coil	R/B RCA	E	50'-2"	FA2-409-01	above flood elevation	0.58	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 11 of 20)

Item No.	Equipment Tag	Description	Location						Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor			
148~~6~~	VRS-MCL-551C	C-Penetration Area Air Handling Unit Cooling Coil	R/B RCA	W	50'-2"	FA2-410-01	above flood elevation		0.76	
149~~7~~	VRS-MCL-551D	D-Penetration Area Air Handling Unit Cooling Coil	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation		0.76	
150~~48~~	VRS-MEH-551A	A-Penetration Area Air Handling Unit Electric Heating Coil	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation		0.58	
151~~49~~	VRS-MEH-551B	B-Penetration Area Air Handling Unit Electric Heating Coil	R/B RCA	E	50'-2"	FA2-409-01	above flood elevation		0.58	
152~~0~~	VRS-MEH-551C	C-Penetration Area Air Handling Unit Electric Heating Coil	R/B RCA	W	50'-2"	FA2-410-01	above flood elevation		0.76	
153~~1~~	VRS-MEH-551D	D-Penetration Area Air Handling Unit Electric Heating Coil	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation		0.76	
154~~2~~	VCS-AOV-304	Air Operated Valve	R/B RCA	E	76'-5"	FA2-~~117-34~~409-02	above flood elevation		0.84	
155~~3~~	VCS-AOV-307	Air Operated Valve	R/B RCA	W	76'-5"	FA2-~~117-40~~511-01	above flood elevation		0.99	
156~~4~~	VCS-AOV-354	Air Operated Valve	R/B RCA	E	76'-5"	FA2-~~117-34~~409-02	above flood elevation		0.84	
157~~5~~	VCS-AOV-357	Air Operated Valve	R/B RCA	W	76'-5"	FA2-~~117-40~~511-01	above flood elevation		0.99	
158~~6~~	VAS-AOD-501A	Air Operated Damper	R/B RCA	E	25'-3"	FA2-~~117-90~~209-04	above flood elevation		0.53	
159~~7~~	VAS-AOD-501B	Air Operated Damper	R/B RCA	W	50'-2"	FA2-~~117-43~~418-01	above flood elevation		0.76	
160~~58~~	VAS-AOD-502A	Air Operated Damper	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation		0.53	
161~~59~~	VAS-AOD-502B	Air Operated Damper	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation		0.76	
162~~0~~	VAS-AOD-503A	Air Operated Damper	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation		0.53	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 12 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
163~~1~~	VAS-AOD-503B	Air Operated Damper	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
164~~2~~	VAS-AOD-504A	Air Operated Damper	R/B RCA	E	25'-3"	FA2-~~117-90~~209-04	above flood elevation	0.53	
165~~3~~	VAS-AOD-504B	Air Operated Damper	R/B RCA	W	50'-2"	FA2-~~117-43~~418-01	above flood elevation	0.76	
166~~4~~	VAS-AOD-505A	Air Operated Damper	R/B RCA	E	3'-7"	FA2-~~117-08~~209-03	above flood elevation	0.69	
167~~5~~	VAS-AOD-505B	Air Operated Damper	R/B RCA	E	3'-7"	FA2-151-04	above flood elevation	0.69	
168~~6~~	VAS-AOD-505C	Air Operated Damper	R/B RCA	W	3'-7"	FA2-152-04	above flood elevation	0.88	
169~~7~~	VAS-AOD-505D	Air Operated Damper	R/B RCA	W	3'-7"	FA2-~~117-07~~128-02	above flood elevation	0.88	
170~~68~~	VAS-AOD-506A	Air Operated Damper	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
171~~69~~	VAS-AOD-506B	Air Operated Damper	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1
172~~0~~	VAS-AOD-506C	Air Operated Damper	R/B RCA	W	3'-7"	FA2-152-03	N/A	-	1
173~~1~~	VAS-AOD-506D	Air Operated Damper	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
174~~2~~	VAS-AOD-507A	Air Operated Damper	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
175~~3~~	VAS-AOD-507B	Air Operated Damper	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1
176~~4~~	VAS-AOD-507C	Air Operated Damper	R/B RCA	W	3'-7"	FA2-~~152-03~~209-03	N/A	-	1
177~~5~~	VAS-AOD-507D	Air Operated Damper	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
178~~6~~	VAS-AOD-508A	Air Operated Damper	R/B RCA	E	3'-7"	FA2-117-08	above flood elevation	0.69	
179~~7~~	VAS-AOD-508B	Air Operated Damper	R/B RCA	E	3'-7"	FA2-151-04	above flood elevation	0.69	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 13 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
180~~78~~	VAS-AOD-508C	Air Operated Damper	R/B RCA	W	3'-7"	FA2-152-04	above flood elevation	0.88	
181~~79~~	VAS-AOD-508D	Air Operated Damper	R/B RCA	W	3'-7"	FA2-~~117-07~~128-02	above flood elevation	0.88	
182~~0~~	VAS-AOD-511	Air Operated Damper	R/B RCA	W	76'-5"	FA2-~~117-44~~210-21	above flood elevation	0.99	
183~~1~~	VAS-AOD-512	Air Operated Damper	R/B RCA	W	76'-5"	FA2-~~117-44~~210-21	above flood elevation	0.99	
184~~2~~	VWS-TMV-304	Chilled Water Control Valve	R/B RCA	E	3'-7"	FA2-~~117-05~~154-04	N/A	-	1
185~~3~~	VWS-TMV-314	Chilled Water Control Valve	R/B RCA	E	3'-7"	FA2-151-02	N/A	-	1
186~~4~~	VWS-TMV-324	Chilled Water Control Valve	R/B RCA	W	3'-7"	FA2-152-02	N/A	-	1
187~~5~~	VWS-TMV-334	Chilled Water Control Valve	R/B RCA	W	3'-7"	FA2-153-04	N/A	-	1
188~~6~~	VWS-TMV-602A	Chilled Water Control Valve	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
189~~7~~	VWS-TMV-602B	Chilled Water Control Valve	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
190~~88~~	VWS-TMV-612A	Chilled Water Control Valve	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
191~~89~~	VWS-TMV-612B	Chilled Water Control Valve	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
192~~0~~	VWS-TMV-622	Chilled Water Control Valve	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
193~~1~~	VWS-TMV-632	Chilled Water Control Valve	R/B RCA	E	50'-2"	FA2-409-01	above flood elevation	0.58	
194~~2~~	VWS-TMV-642	Chilled Water Control Valve	R/B RCA	W	50'-2"	FA2-410-01	above flood elevation	0.76	
195~~3~~	VWS-TMV-652	Chilled Water Control Valve	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
196~~4~~	VWS-MOV-403	Motor Operated Valve	R/B RCA	W	76'-5"	FA2-~~117-40~~511-01	above flood elevation	0.99	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 14 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
197~~5~~	VWS-MOV-407	Motor Operated Valve	R/B RCA	W	76'-5"	FA2-~~117-40~~511-01	above flood elevation	0.99	
198~~6~~	SRPP-A	Source Range Neutron Flux Preamplifier Panel (Train A)	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
199~~7~~	SRPP-D	Source Range Neutron Flux Preamplifier Panel (Train D)	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
200~~198~~	WRPP-A	Wide Range Neutron Flux Preamplifier Panel (Train A)	R/B RCA	E	50'-2"	FA2-408-01	above flood elevation	0.58	
201~~199~~	WRPP-D	Wide Range Neutron Flux Preamplifier Panel (Train D)	R/B RCA	W	50'-2"	FA2-411-01	above flood elevation	0.76	
202~~0~~	CVS-FT-128	Primary Makeup Water Supply Flow	R/B RCA	W	25'-3"	FA2-~~117-42~~209-05	above flood elevation	0.69	
203~~1~~	CVS-FT-129	Primary Makeup Water Supply Flow	R/B RCA	W	25'-3"	FA2-~~117-42~~209-05	above flood elevation	0.69	
204~~2~~	SIS-FT-062	A - Safety Injection Pump Discharge Flow	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
205~~3~~	SIS-FT-063	B - Safety Injection Pump Discharge Flow	R/B RCA	E	-26'-4"	FA2-114-03	N/A		1
206~~4~~	SIS-FT-064	C - Safety Injection Pump Discharge Flow	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1
207~~5~~	SIS-FT-065	D - Safety Injection Pump Discharge Flow	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
208~~6~~	SIS-PT-060	A - Safety Injection Pump Suction Pressure	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
209~~7~~	SIS-PT-061	B - Safety Injection Pump Suction Pressure	R/B RCA	E	-26'-4"	FA2-114-03	N/A		1

Table 3K-2 R/B RCA Components Protected From Internal Flooding
(Sheet 15 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
210~~08~~	SIS-PT-062	C - Safety Injection Pump Suction Pressure	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1
211~~09~~	SIS-PT-063	D - Safety Injection Pump Suction Pressure	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
212~~0~~	SIS-PT-064	A - Safety Injection Pump Discharge Pressure	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
213~~1~~	SIS-PT-065	B - Safety Injection Pump Discharge Pressure	R/B RCA	E	-26'-4"	FA2-114-03	N/A		1
214~~2~~	SIS-PT-066	C - Safety Injection Pump Discharge Pressure	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1
215~~3~~	SIS-PT-067	D - Safety Injection Pump Discharge Pressure	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
216~~4~~	RHS-FT-011	A - Containment Spray / Residual Heat Removal Pump Discharge Flow	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
217~~5~~	RHS-FT-014	A - Containment Spray / Residual Heat Removal Pump Minimum Flow	R/B RCA	E	3'-7"	FA2-~~117-08~~209-03	above flood elevation	0.69	
218~~6~~	RHS-FT-021	B - Containment Spray / Residual Heat Removal Pump Discharge Flow	R/B RCA	E	-26'-4"	FA2-114-03	N/A		
219~~7~~	RHS-FT-024	B - Containment Spray / Residual Heat Removal Pump Minimum Flow	R/B RCA	E	3'-7"	FA2-151-04	above flood elevation	0.69	
220~~18~~	RHS-FT-031	C - Containment Spray / Residual Heat Removal Pump Discharge Flow	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1

Table 3K-2 R/B RCA Components Protected From Internal Flooding
(Sheet 16 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
211~~19~~	RHS-FT-034	C - Containment Spray / Residual Heat Removal Pump Minimum Flow	R/B RCA	W	3'-7"	FA2-152-04	above flood elevation	0.88	
222~~0~~	RHS-FT-041	D - Containment Spray / Residual Heat Removal Pump Discharge Flow	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
223~~1~~	RHS-FT-044	D - Containment Spray / Residual Heat Removal Pump Minimum Flow	R/B RCA	W	3'-7"	FA2-~~117-07~~128-02	above flood elevation	0.88	
224~~2~~	RHS-PT-010	A - Containment Spray / Residual Heat Removal Pump Suction Pressure	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
225~~3~~	RHS-PT-011	A - Containment Spray / Residual Heat Removal Pump Discharge Pressure	R/B RCA	E	-26'-4"	FA2-113-03	N/A		1
226~~4~~	RHS-PT-020	B - Containment Spray / Residual Heat Removal Pump Suction Pressure	R/B RCA	E	-26'-4"	FA2-114-03	N/A		1
227~~5~~	RHS-PT-021	B - Containment Spray / Residual Heat Removal Pump Discharge Pressure	R/B RCA	E	-26'-4"	FA2-114-03	N/A		1
228~~6~~	RHS-PT-~~020~~030	C - Containment Spray / Residual Heat Removal Pump Suction Pressure	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1
229~~7~~	RHS-PT-~~021~~031	C - Containment Spray / Residual Heat Removal Pump Discharge Pressure	R/B RCA	W	-26'-4"	FA2-115-03	N/A		1

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 17 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
230~~28~~	RHS-PT-040	D - Containment Spray / Residual Heat Removal Pump Suction Pressure	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
231~~29~~	RHS-PT-041	D - Containment Spray / Residual Heat Removal Pump Discharge Pressure	R/B RCA	W	-26'-4"	FA2-116-03	N/A		1
232~~0~~	CSS-PT-010	Containment Pressure	R/B RCA	E	~~25'-3"~~76'-5"	FA2-~~151-05~~506-01	above flood elevation	~~0.53~~0.84	
233~~1~~	CSS-PT-011	Containment Pressure	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
234~~2~~	CSS-PT-012	Containment Pressure	R/B RCA	W	~~76'-5"~~25'-3"	FA2-~~117-35~~152-05	above flood elevation	~~0.88~~0.69	
235~~3~~	CSS-PT-013	Containment Pressure	R/B RCA	W	76'-5"	FA2-~~117-35~~410-02	above flood elevation	~~0.88~~0.99	
236~~4~~	RHS-TE-014	A - Containment Spray / Residual Heat Removal Heat Exchanger Outlet Temperature	R/B RCA	E	3'-7"	FA2-~~120-02~~154-03	N/A	-	1
237~~5~~	RHS-TE-024	B - Containment Spray / Residual Heat Removal Heat Exchanger Outlet Temperature	R/B RCA	E	3'-7"	FA2-151-~~01~~03	N/A	-	1
238~~6~~	RHS-TE-~~024~~034	C - Containment Spray / Residual Heat Removal Heat Exchanger Outlet Temperature	R/B RCA	W	3'-7"	FA2-152-~~01~~03	N/A	-	1
239~~7~~	RHS-TE-~~034~~044	D - Containment Spray / Residual Heat Removal Heat Exchanger Outlet Temperature	R/B RCA	W	3'-7"	FA2-153-~~01~~03	N/A	-	1
240~~38~~	VRS-TS-621	A - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 18 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
241~~39~~	VRS-TS-624	A - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
242~~0~~	VRS-TS-625	A - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-~~120-06~~154-05	above flood elevation	0.53	
243~~1~~	VRS-TS-631	B - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
244~~2~~	VRS-TS-634	B - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
245~~3~~	VRS-TS-635	B - Penetration Area Temperature	R/B RCA	E	25'-3"	FA2-151-05	above flood elevation	0.53	
246~~4~~	VRS-TS-641	C - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
247~~5~~	VRS-TS-644	C - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
248~~6~~	VRS-TS-645	C - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-152-05	above flood elevation	0.69	
249~~7~~	VRS-TS-651	D - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
250~~48~~	VRS-TS-654	D - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
251~~49~~	VRS-TS-655	D - Penetration Area Temperature	R/B RCA	W	25'-3"	FA2-153-05	above flood elevation	0.69	
252~~0~~	VRS-TS-306	A - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
253~~1~~	VRS-TS-307	A - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
254~~2~~	VRS-TS-305	A - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-~~120-04~~154-03	N/A	-	1
255~~3~~	VRS-TS-316	B - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1

Table 3K-2 R/B RCA Components Protected From Internal Flooding
(Sheet 19 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
256~~4~~	VRS-TS-317	B - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1
257~~5~~	VRS-TS-315	B - Safeguard Component Area Temperature	R/B RCA	E	3'-7"	FA2-151-03	N/A	-	1
258~~6~~	VRS-TS-~~526~~326	C - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-152-03	N/A	-	1
259~~7~~	VRS-TS-327	C - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-152-03	N/A	-	1
260~~58~~	VRS-TS-325	C - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-152-03	N/A	-	1
261~~59~~	VRS-TS-336	D - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
262~~0~~	VRS-TS-337	D - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
263~~1~~	VRS-TS-335	D - Safeguard Component Area Temperature	R/B RCA	W	3'-7"	FA2-153-03	N/A	-	1
264~~2~~	VRS-TS-601	A - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
265~~3~~	VRS-TS-604	A - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
266~~4~~	VRS-TS-605	A - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	E	50'-2"	FA2-~~117-32~~416-01	above flood elevation	0.58	
267~~5~~	VRS-TS-611	B - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	
268~~6~~	VRS-TS-614	B - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	

Table 3K-2 R/B RCA Components Protected From Internal Flooding (Sheet 20 of 20)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
269~~7~~	VRS-TS-615	B - Annulus Emergency Exhaust Filtration Unit Area Temperature	R/B RCA	W	50'-2"	FA2-~~117-29~~417-01	above flood elevation	0.76	

Note:

These components are protected by water-tight door and floor drain isolation valve against in-flow of flooding occurring outside of compartment. In addition, these components are not required to be protected against flooding occurring inside the compartment due to redundancy of other trains/components.

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 1 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
1	EFS-MPP-001A	A-Emergency Feedwater Pump	R/B NRCA	E	-26'-4"	FA2-102-01	above flood elevation	0.45	
2	EFS-MPP-001B	B-Emergency Feedwater Pump	R/B NRCA	E	-26'-4"	FA2-103-01	above flood elevation	0.45	
3	EFS-MPP-001C	C-Emergency Feedwater Pump	R/B NRCA	W	-26'-4"	FA2-109-01	above flood elevation	0.60	
4	EFS-MPP-001D	D-Emergency Feedwater Pump	R/B NRCA	W	-26'-4"	FA2-108-01	above flood elevation	0.60	
5	EFS-MPT-001A	A-Emergency Feedwater Pit	R/B NRCA	E	76'-5"	FA2-501-02	0	1.50	2
6	EFS-MPK-001B	B-Emergency Feedwater Pit	R/B NRCA	W	76'-5"	FA2-~~501-08~~512-01	0	1.24	2
7	EFS-MOV-014A	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-102-01	above flood elevation	0.45	
8	EFS-MOV-014B	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-103-01	above flood elevation	0.45	
9	EFS-MOV-014C	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-109-01	above flood elevation	0.60	
10	EFS-MOV-014D	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-108-01	above flood elevation	0.60	
11	EFS-MOV-017A	A-Emergency Feedwater Control Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
12	EFS-MOV-017B	B-Emergency Feedwater Control Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
13	EFS-MOV-017C	C-Emergency Feedwater Control Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
14	EFS-MOV-017D	D-Emergency Feedwater Control Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
15	EFS-MOV-019A	A-Emergency Feedwater Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 2 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
16	EFS-MOV-019B	B-Emergency Feedwater Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
17	EFS-MOV-019C	C-Emergency Feedwater Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
18	EFS-MOV-019D	D-Emergency Feedwater Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
19	EFS-MOV-101A	A-Emergency Feedwater Pump A-Main Steam Line Steam Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
20	EFS-MOV-101B	A-Emergency Feedwater Pump B-Main Steam Line Steam Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
21	EFS-MOV-101C	D-Emergency Feedwater Pump C-Main Steam Line Steam Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
22	EFS-MOV-101D	D-Emergency Feedwater Pump D-Main Steam Line Steam Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
23	EFS-MOV-103A, EFS-MOV-103B	A-Emergency Feedwater Pump Actuation Valve on A-steam supply line, A-Emergency Feedwater Pump Actuation Valve on B-steam supply line	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
24	EFS-MOV-103C, EFS-MOV-103D	~~B~~D-Emergency Feedwater Pump Actuation Valve on C-steam supply line, D-Emergency Feedwater Pump Actuation Valve on D-steam supply line	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
25	FWS-SMV~~LV~~-512A	A-Main Feedwater Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	below flood elevation	4.6	3
26	FWS-SMV~~LV~~-512B	B-Main Feedwater Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	below flood elevation	4.6	3

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 3 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
27	FWS-SMV~~LV~~-512C	C-Main Feedwater Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	below flood elevation	4.6	3
28	FWS-SMV~~LV~~-512D	D-Main Feedwater Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	below flood elevation	4.6	3
29	MSS-~~VL~~SRV-509A	A1-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
30	MSS-~~VL~~SRV-510A	A2-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
31	MSS-~~VL~~SRV-511A	A3-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
32	MSS-~~VL~~SRV-512A	A4-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
33	MSS-~~VL~~SRV-513A	A5-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
34	MSS-~~VL~~SRV-514A	A6-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
35	MSS-~~VL~~SRV-509B	B1-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
36	MSS-~~VL~~SRV-510B	B2-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
37	MSS-~~VL~~SRV-511B	B3-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
38	MSS-~~VL~~SRV-512B	B4-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
39	MSS-~~VL~~SRV-513B	B5-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
40	MSS-~~VL~~SRV-514B	B6-Main Steam Safety Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
41	MSS-~~VL~~SRV-509C	C1-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
42	MSS-~~VL~~SRV-510C	C2-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
43	MSS-~~VL~~SRV-511C	C3-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 4 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
44	MSS-~~VL~~SRV-512C	C4-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
45	MSS-~~VL~~SRV-513C	C5-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
46	MSS-~~VL~~SRV-514C	C6-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
47	MSS-~~VL~~SRV-509D	D1-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
48	MSS-~~VL~~SRV-510D	D2-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
49	MSS-~~VL~~SRV-511D	D3-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
50	MSS-~~VL~~SRV-512D	D4-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
51	MSS-~~VL~~SRV-513D	D5-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
52	MSS-~~VL~~SRV-514D	D6-Main Steam Safety Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
53	MSS-MOV-507A	A-Main Steam Relief Valve Block Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
54	MSS-MOV-507B	B-Main Steam Relief Valve Block Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
55	MSS-MOV-507C	C-Main Steam Relief Valve Block Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
56	MSS-MOV-507D	D-Main Steam Relief Valve Block Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
57	MSS-MOV-508A	A-Main Steam Depressurization Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
58	MSS-MOV-508B	B-Main Steam Depressurization Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 5 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
59	MSS-MOV-508C	C-Main Steam Depressurization Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
60	MSS-MOV-508D	D-Main Steam Depressurization Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
61	MSS-~~AOV~~SMV-515A	A-Main Steam Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
62	MSS-~~AOV~~SMV-515B	B-Main Steam Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
63	MSS-~~AOV~~SMV-515C	C-Main Steam Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
64	MSS-~~AOV~~SMV-515D	D-Main Steam Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
65	MSS-HCV-565	A-Main Steam Bypass Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
66	MSS-HCV-575	B-Main Steam Bypass Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
67	MSS-HCV-585	C-Main Steam Bypass Isolation Valve Hand Control Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
68	MSS-HCV-595	D-Main Steam Bypass Isolation Valve Hand Control Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
69	MSS-PCV-515	A-Main Steam Relief Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
70	MSS-PCV-525	B-Main Steam Relief Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
71	MSS-PCV-535	C-Main Steam Relief Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
72	MSS-PCV-545	D-Main Steam Relief Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
73	MSS-MOV-701A	A-Main Steam Drain Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 6 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
74	MSS-MOV-701B	B-Main Steam Drain Isolation Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
75	MSS-MOV-701C	C-Main Steam Drain Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
76	MSS-MOV-701D	D-Main Steam Drain Isolation Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
77	NCS-MPP-001A	A-Component Cooling Water Pump	R/B NRCA	E	-26'-4"	FA2-104-01	above flood elevation	0.45	
78	NCS-MPP-001B	B-Component Cooling Water Pump	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
79	NCS-MPP-001C	C-Component Cooling Water Pump	R/B NRCA	W	-26'-4"	FA2-106-01	above flood elevation	0.60	
80	NCS-MPP-001D	D-Component Cooling Water Pump	R/B NRCA	W	-26'-4"	FA2-107-01	above flood elevation	0.60	
81	NCS-MTK-001A	A-Component Cooling Water Surge tank	R/B NRCA	E	101'-0"	FA2-601-01	below flood elevation	1.71	4
82	NCS-MTK-001B	B-Component Cooling Water Surge Tank	R/B NRCA	W	101'-0"	FA2-602-01	below flood elevation	3.08	5
83	NCS-MHX-001A	A-Component Cooling Water Heat Exchanger	R/B NRCA	E	-26'-4"	FA2-104-01	above flood elevation	0.45	
84	NCS-MHX-001B	B-Component Cooling Water Heat Exchanger	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
85	NCS-MHX-001C	C-Component Cooling Water Heat Exchanger	R/B NRCA	W	-26'-4"	FA2-106-01	above flood elevation	0.60	
86	NCS-MHX-001D	D-Component Cooling Water Heat Exchanger	R/B NRCA	W	-26'-4"	FA2-107-01	above flood elevation	0.60	
87	NCS-~~VLV~~SRV-003A	Safety Valve	R/B NRCA	E	101'-0"	FA2-601-01	above flood elevation	1.71	
88	NCS-~~VLV~~SRV-003B	Safety Valve	R/B NRCA	W	101'-0"	FA2-602-01	above flood elevation	3.08	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 7 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
89	NCS-MOV-007A	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-104-01	above flood elevation	0.45	
90	NCS-MOV-007B	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
91	NCS-MOV-020A	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-104-01	above flood elevation	0.45	
92	NCS-MOV-020B	Motor Operated Valve	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
93	NCS-VLV-035A	Safety Valve	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
94	NCS-VLV-035B	Safety Valve	R/B NRCA	~~E~~W	-26'-4"	FA2-106-01	above flood elevation	0.60	
95	NCS-RCV-056A	Radiation Control Valve	R/B NRCA	E	101'-0"	FA2-601-01	above flood elevation	1.71	
96	NCS-LCV-~~1200~~010	Level Control Valve	R/B NRCA	E	101'-0"	FA2-60~~1~~3-01	above flood elevation	1.71	
97	NCS-MOV-007C	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-106-01	above flood elevation	0.60	
98	NCS-MOV-007D	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-107-01	above flood elevation	0.60	
99	NCS-MOV-020C	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-106-01	above flood elevation	0.60	
100	NCS-MOV-020D	Motor Operated Valve	R/B NRCA	W	-26'-4"	FA2-107-01	above flood elevation	0.60	
101	NCS-RCV-056B	Radiation Control Valve	R/B NRCA	W	101'-0"	FA2-602-01	above flood elevation	3.08	
102	NCS-LCV-020	Level Control Valve	R/B NRCA	W	101'-0"	FA2-60~~2~~4-01	below flood elevation	3.08	7
103	NCS-PCV-012	Pressure Control Valve	R/B NRCA	E	101'-0"	FA2-601-01	above flood elevation	1.71	
104	NCS-PCV-022	Pressure Control Valve	R/B NRCA	W	101'-0"	FA2-602-01	above flood elevation	3.08	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 8 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
105	EWS-SS~~R~~T-003A	A-Component Cooling Water Heat Exchanger Inlet Strainer	R/B NRCA	E	-26'-4"	FA2-104-01	above flood elevation	0.45	
106	EWS-SS~~R~~T-003B	B-Component Cooling Water Heat Exchanger Inlet Strainer	R/B NRCA	E	-26'-4"	FA2-105-01	above flood elevation	0.45	
107	EWS-SS~~R~~T-003C	C-Component Cooling Water Heat Exchanger Inlet Strainer	R/B NRCA	W	-26'-4"	FA2-106-01	above flood elevation	0.60	
108	EWS-SS~~R~~T-003D	D-Component Cooling Water Heat Exchanger Inlet Strainer	R/B NRCA	W	-26'-4"	FA2-107-01	above flood elevation	0.60	
109	SGS-AOV-001A	Air Operated Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
110	SGS-AOV-001B	Air Operated Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
111	SGS-AOV-001C	Air Operated Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
112	SGS-AOV-001D	Air Operated Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
113	SGS-AOV-002A	Air Operated Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
114	SGS-AOV-002B	Air Operated Valve	R/B NRCA	E	65'-0"	FA2-414-01	above flood elevation	4.6	
115	SGS-AOV-002C	Air Operated Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
116	SGS-AOV-002D	Air Operated Valve	R/B NRCA	W	65'-0"	FA2-415-01	above flood elevation	4.6	
117	VRS-MAH-101A	A-Main Control Room Air Handling Unit	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
118	VRS-MAH-101B	B-Main Control Room Air Handling Unit	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
119	VRS-MAH-101C	C-Main Control Room Air Handling Unit	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 10 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
133	VRS-MFU-111A	A-Main Control Room Emergency Filtration Unit	R/B NRCA	E	50'-2"	FA2-405-01	above flood elevation	0.87	
134	VRS-MFU-111B	B-Main Control Room Emergency Filtration Unit	R/B NRCA	W	50'-2"	FA2-406-01	above flood elevation	0.86	
135	VRS-MFN-111A	A-Main Control Room Emergency Filtration Unit Fan	R/B NRCA	E	50'-2"	FA2-405-01	above flood elevation	0.87	
136	VRS-MFN-111B	B-Main Control Room Emergency Filtration Unit Fan	R/B NRCA	W	50'-2"	FA2-406-01	above flood elevation	0.86	
137	VRS-MEH-111A	A-Main Control Room Emergency Filtration Unit Electric Heating Coil	R/B NRCA	E	50'-2"	FA2-405-01	above flood elevation	0.87	
138	VRS-MEH-111B	B-Main Control Room Emergency Filtration Unit Electric Heating Coil	R/B NRCA	W	50'-2"	FA2-406-01	above flood elevation	0.86	
139	VRS-EHD-101A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-~~407-04~~420-01	above flood elevation	0.87	
140	VRS-EHD-101B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-~~407~~423-01	above flood elevation	0.86	
141	VRS-EHD-102A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-~~407-04~~420-01	above flood elevation	0.87	
142	VRS-EHD-102B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-~~407~~423-01	above flood elevation	0.86	
143	VRS-AOD-103A	Air Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
144	VRS-AOD-103B	Air Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
145	VRS-EHD-104A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
146	VRS-EHD-104B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
147	VRS-EHD-105A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 11 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
148	VRS-EHD-105B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
149	VRS-EHD-105C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
150	VRS-EHD-105D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
151	VRS-EHD-106A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
152	VRS-EHD-106B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
153	VRS-EHD-106C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
154	VRS-EHD-106D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
155	VRS-EHD-107A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
156	VRS-EHD-107B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
157	VRS-MOD-111A	Motor Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
158	VRS-MOD-111B	Motor Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
159	VRS-MOD-112A	Motor Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	
160	VRS-MOD-112B	Motor Operated Damper	R/B NRCA	W	50'-2"	FA2-413-01	above flood elevation	0.86	
161	VRS-MOD-113A	Motor Operated Damper	R/B NRCA	E	50'-2"	FA2-405-01	above flood elevation	0.87	
162	VRS-MOD-113B	Motor Operated Damper	R/B NRCA	W	50'-2"	FA2-406-01	above flood elevation	0.86	
163	VRS-AOD-121	Air Operated Damper	R/B NRCA	E	26'-11"	FA2-308-02	N/A	-	6
164	VRS-AOD-122	Air Operated Damper	R/B NRCA	E	50'-2"	FA2-412-01	above flood elevation	0.87	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 13 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
178	VRS-MFN-202D	D-Class 1E Electrical Room Return Air Fan	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
179	VRS-MCL-201A	A-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
180	VRS-MCL-201B	B-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
181	VRS-MCL-201C	C-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
182	VRS-MCL-201D	D-Class 1E Electrical Room Air Handling Unit Cooling Coil	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
183	VRS-MEH-201A	A-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
184	VRS-MEH-201B	B-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
185	VRS-MEH-201C	C-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
186	VRS-MEH-201D	D-Class 1E Electrical Room Air Handling Unit Electric Heating Coil	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
187	VRS-MEH-202A	A-Class 1E I&C Room In-Duct Heater	R/B NRCA	E	25’-3”	FA2-304-01	NA	-	6
188	VRS-MEH-202B	B-Class 1E I&C Room In-Duct Heater	R/B NRCA	E	25’-3”	FA2-307-01	NA	-	6
189	VRS-MEH-202C	C-Class 1E I&C Room In-Duct Heater	R/B NRCA	W	25’-3”	FA2-312-01	NA	-	6
190	VRS-MEH-202D	D-Class 1E I&C Room In-Duct Heater	R/B NRCA	W	25’-3”	FA2-309-01	NA	-	6

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 14 of 30)

Item No.	Equipment Tag	Description	Location: Building	Location: Side	Location: Floor Elevation	Location: Fire Zone No.	Location: Location Elevation above Floor	Flood Elevation above Floor [ft]	Notes
191	VRS-MEH-203A	A - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
192	VRS-MEH-203B	B - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
193	VRS-MEH-203C	C - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
194	VRS-MEH-203D	D - Class 1E Electrical Room MCR HVAC Equipment Room In-Duct Heater	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
195	VRS-MEH-211A	A - Remote Shutdown Console Room In-Duct Heater	R/B NRCA	E	76'-5"	FA2-504-01	NA	-	6
196	VRS-MEH-211B	B - Remote Shutdown Console Room In-Duct Heater	R/B NRCA	E	76'-5"	FA2-504-01	NA	-	6
197	VRS-MEH-204A	A - Class 1E Battery Room In-Duct Heater	R/B NRCA	E	-26'-4"	FA3-115-01	above flood elevation	0.45	
198	VRS-MEH-204B	B - Class 1E Battery Room In-Duct Heater	R/B NRCA	E	-26'-4"	FA3-116-01	above flood elevation	0.45	
199	VRS-MEH-204C	C - Class 1E Battery Room In-Duct Heater	R/B NRCA	W	-26'-4"	FA3-120-01	above flood elevation	0.60	
200	VRS-MEH-204D	D - Class 1E Battery Room In-Duct Heater	R/B NRCA	W	-26'-4"	FA3-121-01	above flood elevation	0.60	

Table 3K-3 R/B NRCA Components Protected From Internal Flooding (Sheet 15 of 30)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
201	VRS-EHD-201A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
202	VRS-EHD-201B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
203	VRS-EHD-201C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
204	VRS-EHD-201D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
205	VRS-EHD-202A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
206	VRS-EHD-202B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
207	VRS-EHD-202C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
208	VRS-EHD-202D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
209	VRS-EHD-203A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
210	VRS-EHD-203B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
211	VRS-EHD-203C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
212	VRS-EHD-203D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
213	VRS-EHD-204A	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	
214	VRS-EHD-204B	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	E	50'-2"	FA2-401-01	above flood elevation	0.87	
215	VRS-EHD-204C	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-403-01	above flood elevation	0.86	
216	VRS-EHD-204D	Electro Hydraulic~~Motor~~ Operated Damper	R/B NRCA	W	50'-2"	FA2-404-01	above flood elevation	0.86	
217	VRS-AOD-205A	Air Operated Damper	R/B NRCA	E	50'-2"	FA2-402-01	above flood elevation	0.87	

Table 3K-4 PS/B Components Protected From Internal Flooding (Sheet 1 of 7)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
1	VRS-~~R~~MFN-251A	A-Class 1E Battery Room Exhaust Fan	PS/B	E	3'-7"	FA3-104-03~~4~~	N/A	-	1
2	VRS-~~R~~MFN-251B	B-Class 1E Battery Room Exhaust Fan	PS/B	E	3'-7"	FA3-103-03	N/A	-	1
3	VRS-~~R~~MFN-251C	C-Class 1E Battery Room Exhaust Fan	PS/B	W	3'-7"	FA3-109-03	N/A	-	1
4	VRS-~~R~~MFN-251D	D-Class 1E Battery Room Exhaust Fan	PS/B	W	3'-7"	FA3-111-03~~4~~	N/A	-	1
5	VRS-EHD-251A	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	E	3'-7"	FA3-104-03~~4~~	N/A	-	1
6	VRS-EHD-251B	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	E	3'-7"	FA3-103-03	N/A	-	1
7	VRS-EHD-251C	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	W	3'-7"	FA3-109-03	N/A	-	1
8	VRS-EHD-251D	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	W	3'-7"	FA3-111-03~~4~~	N/A	-	1
9	VRS-EHD-252A	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	E	3'-7"	FA3-104-03~~4~~	N/A	-	1
10	VRS-EHD-252B	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	E	3'-7"	FA3-103-03	N/A	-	1
11	VRS-EHD-252C	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	W	3'-7"	FA3-109-03	N/A	-	1
12	VRS-EHD-252D	Electro Hydraulic~~Motor~~ Operated Damper	PS/B	W	3'-7"	FA3-111-03~~4~~	N/A	-	1
13	VRS-MAH-511A	A-Essential Chiller Unit Area Air Handling Unit	PS/B	E	-26'-4"	FA3-101-01	above flood elevation	0.45	2
14	VRS-MAH-511B	B-Essential Chiller Unit Area Air Handling Unit	PS/B	E	-26'-4"	FA3-102-01	above flood elevation	0.45	2

Table 3K-4 PS/B Components Protected From Internal Flooding (Sheet 4 of 7)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
43	VWS-TMV-562	Chilled Water Control Valve	PS/B	W	-26'-4"	FA3-108-01	above flood elevation	0.60	2
44	VWS-TMV-572	Chilled Water Control Valve	PS/B	W	-26'-4"	FA3-110-01	above flood elevation	0.60	2
45	VWS-~~VLV~~SRV-253A	Safety Valve	PS/B	E	-26'-4"	FA3-101-01	above flood elevation	0.45	2
46	VWS-~~VLV~~SRV-253B	Safety Valve	PS/B	E	-26'-4"	FA3-102-01	above flood elevation	0.45	2
47	VWS-~~VLV~~SRV-253C	Safety Valve	PS/B	W	-26'-4"	FA3-108-01	above flood elevation	0.60	2
48	VWS-~~VLV~~SRV-253D	Safety Valve	PS/B	W	-26'-4"	FA3-110-01	above flood elevation	0.60	2
49	A-EGTG	A-Class 1E Gas Turbine Generator	PS/B	E	3'-7"	FA3-104-03~~4~~	N/A	-	1
50	B-EGTG	B-Class 1E Gas Turbine Generator	PS/B	E	3'-7"	FA3-103-03	N/A	-	1
51	C-EGTG	C-Class 1E Gas Turbine Generator	PS/B	W	3'-7"	FA3-109-03	N/A	-	1
52	D-EGTG	D-Class 1E Gas Turbine Generator	PS/B	W	3'-7"	FA3-111-03~~4~~	N/A	-	1
53	BCP-A	A-Class 1E Battery Charger	PS/B	E	-14'-2"	FA3-117-01	N/A	-	1
54	DCC-A	A-Class 1E DC Switchboard	PS/B	E	-14'-2"	FA3-117-01	N/A	-	1
55	DCC-A1	A1-Class 1E DC Switchboard	PS/B	E	-14'-2"	FA3-117-01	N/A	-	1
56	BCP-B	B-Class 1E Battery Charger	PS/B	E	-14'-2"	FA3-118-01	N/A	-	1
57	DCC-B	B-Class 1E DC Switchboard	PS/B	E	-14'-2"	FA3-118-01	N/A	-	1
58	BCP-C	C-Class 1E Battery Charger	PS/B	W	-14'-2"	FA3-122-01	N/A	-	1
59	DCC-C	C-Class 1E DC Switchboard	PS/B	W	-14'-2"	FA3-122-01	N/A	-	1

Table 3K-4 PS/B Components Protected From Internal Flooding (Sheet 5 of 7)

Item No.	Equipment Tag	Description	Location						Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor			
60	BCP-D	D-Class 1E Battery Charger	PS/B	W	-14'-2"	FA3-123-01	N/A		-	1
61	DCC-D	D-Class 1E DC Switchboard	PS/B	W	-14'-2"	FA3-123-01	N/A		-	1
62	DCC-D1	D1-Class 1E DC Switchboard	PS/B	W	-14'-2"	FA3-123-01	N/A		-	1
63	VCC-A	A-Ventilation Chiller Control Cabinet	PS/B	E	-26'-4"	FA3-101-01	above flood elevation		0.45	2
64	VCC-B	B-Ventilation Chiller Control Cabinet	PS/B	E	-26'-4"	FA3-102-01	above flood elevation		0.45	2
65	VCC-C	C-Ventilation Chiller Control Cabinet	PS/B	W	-26'-4"	FA3-108-01	above flood elevation		0.60	2
66	VCC-D	D-Ventilation Chiller Control Cabinet	PS/B	W	-26'-4"	FA3-110-01	above flood elevation		0.60	2
67	BAT-A	A-Class 1E Battery	PS/B	E	-26'-4"	FA3-115-01	above flood elevation		0.45	2
68	BAT-B	B-Class 1E Battery	PS/B	E	-26'-4"	FA3-116-01	above flood elevation		0.45	2
69	BAT-C	C-Class 1E Battery	PS/B	W	-26'-4"	FA3-120-01	above flood elevation		0.60	2
70	BAT-D	D-Class 1E Battery	PS/B	W	-26'-4"	FA3-121-01	above flood elevation		0.60	2
71	EPBA	A-Class 1E Gas Turbine Generator Control Board	PS/B	E	3'-7"	FA3-104-034	N/A		-	1
72	EPBB	B-Class 1E Gas Turbine Generator Control Board	PS/B	E	3'-7"	FA3-103-03	N/A		-	1
73	EPBC	C-Class 1E Gas Turbine Generator Control Board	PS/B	W	3'-7"	FA3-109-03	N/A		-	1

Table 3K-4 PS/B Components Protected From Internal Flooding (Sheet 6 of 7)

Item No.	Equipment Tag	Description	Location					Flood Elevation above Floor [ft]	Notes
			Building	Side	Floor Elevation	Fire Zone No.	Location Elevation above Floor		
74	EPBD	D-Class 1E Gas Turbine Generator Control Board	PS/B	W	3'-7"	FA3-111-034	N/A	-	1
75	VRS-TS-541	A - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-101-01	above flood elevation	0.45	2
76	VRS-TS-544	A - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-101-01	above flood elevation	0.45	2
77	VRS-TS-545	A - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-101-01	above flood elevation	0.45	2
78	VRS-TS-551	B - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-102-01	above flood elevation	0.45	2
79	VRS-TS-554	B - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-102-01	above flood elevation	0.45	2
80	VRS-TS-555	B - Essential Chiller Unit Area Temperature	PS/B	E	-26'-4"	FA3-102-01	above flood elevation	0.45	2
81	VRS-TS-561	C - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-108-01	above flood elevation	0.60	2
82	VRS-TS-564	C - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-108-01	above flood elevation	0.60	2
83	VRS-TS-565	C - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-108-01	above flood elevation	0.60	2
84	VRS-TS-571	D - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-110-01	above flood elevation	0.60	2
85	VRS-TS-574	D - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-110-01	above flood elevation	0.60	2
86	VRS-TS-575	D - Essential Chiller Unit Area Temperature	PS/B	W	-26'-4"	FA3-110-01	above flood elevation	0.60	2

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-1 Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-3 Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-4 Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 13'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-5 Location of Watertight Doors and Flood Barrier Walls
R/B Plan View Elevation 25'-3"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-6 Location of Watertight Doors and Flood Barrier Walls
R/B Plan View Elevation 35'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-7 Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 50'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-8 Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 76'-5"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3K-12 Location of Watertight Doors and Flood Barrier Walls PS/Bs Plan View Elevation 3'-7", 24'-2", 39'-6"

Chapter 4

US-APWR DCD Chapter 4 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
4.2-35	4.2.6	Editorial Corrections - The revision number of the Topical/Technical Report described in References is changed to the latest version.
4.4-27	4.4.6.3 The sixth paragraph	Editorial Corrections: Correct the following statement to keep the consistency with the description in Tier 1 Subsection 2.4.3 Loose Parts Monitoring. - Replace ”Two redundant sensors are mounted” with “At least, two redundant sensors are mounted”. - Remove “their associated instrument channels are”
4.6-1	4.6.2	Editorial Corrections - Change “The CRDS takes a part ….” to “ The CRDS is part ….”. - Change “The detail of the ….” to “Detail of the ….”. - Change “….trip system is also performed as described in ….” to “….trip system is described in ….”. - Change “exceeds” to “exceed”.

4.2-4 Reactor Site Criteria, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 100.

4.2-5 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206. Section C.I.4.2.

4.2-6 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.~~0~~2 (Proprietary) and MUAP-07008-NP Rev.~~0~~2 (Non-Proprietary), ~~May 2007~~July 2010.

4.2-7 US-APWR Fuel System Design Evaluation, MUAP-07016-P Rev.~~1~~3 (Proprietary) and MUAP-07016-NP Rev.~~1~~3 (Non-Proprietary), ~~October 2009~~August 2010.

4.2-8 American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III.

4.2-9 W. J. O'Donnel and B. F. Langer, Fatigue Design Basis for Zircaloy Components, Nuclear Science and Engineering 20, pp.1-12, 1964.

4.2-10 US-APWR Fuel System Design Parameters List, MUAP-07018-P Rev.0 (Proprietary) and MUAP-07018-NP Rev.0 (Non-Proprietary), December 2007.

4.2-11 D.Hardie and M.W.Shanahan, Stress Reorientation of Hydrides in Zirconium 2.5 % Niobium, Journal of Nuclear Materials 55, pp.1-13, 1975.

4.2-12 Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix B.

4.2-13 Quality Assurance Requirements for Nuclear Facility Applications, ASME Boiler and Pressure Vessel Code NQA-1 1994.Edition.

4.2-14 FINDS: Mitsubishi PWR Fuel Assemblies Seismic Analysis Code, MUAP-07034-P Rev.~~0~~3 (Proprietary) and MUAP-07034-NP Rev.~~0~~3 (Non-Proprietary), ~~March 2008~~July 2010.

4.2-15 Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads, MUAP-08007-P Rev.~~0~~2 (Proprietary) and MUAP-08007-NP Rev.~~0~~2 (Non-Proprietary), ~~March 2009~~December 2010.

4.4.6.3 Other Monitoring

Other non-safety-related monitoring systems are installed for diagnostics of RV and RCS operating condition.

The loose parts monitoring system provides early detection of loose metallic parts in the RCS. The system is designed to have following characteristics in compliance with Regulatory Guide 1.133 (Reference4.4-19).

The system is an automatic detection system and contains multiple redundant instrumentation channels. Each channel includes a field mounted sensor (piezoelectric accelerometer), preamplifier, signal processing equipment and signal recorder.

The system is designed, as a minimum, to detect a metallic loose part that weighs from 0.25 to 30 lb and impacts with a kinetic energy of 0.5 ft-lb on the inside surface of the RCS pressure boundary within 3 ft of a sensor.

False impact detection, attributable to normal hydraulic, mechanical, and electrical noise, is minimized through the impact detection algorithm:

- Acoustic transients due to known plant operations, such as pump transients, are correlated by time and type of impact and are not alerted as a loose part detection.

- Recurring transients, such as those caused twice within a minute, are more likely to be considered a loose part detection than those that occur at random intervals longer than one minute.

- A floating signal, which is generally attributed to background noise, is used as the approach for setting a reference noise level.

The sensors are strategically located at fixed positions on the RCS to maximize positive impact detection. These positions include the upper and lower plenums of RV and the inlet plenum of each steam generator, which provide broad coverage. At least ~~T~~two redundant sensors are mounted at each location and ~~their associated instrument channels are~~ physically separated from each other. Sensors are carefully mounted to minimize the effect of mechanical force and thermal expansion.

Calibration is performed prior to plant startup. The system is capable of periodic online channel checks, such as real-time audio monitoring and channel functional tests, and offline channel calibrations during refueling outages.

Components within containment are designed and installed to perform their function through and following an earthquake of half the magnitude of a safe-shutdown earthquake (SSE), as well as normal operating radiation, vibration, temperature and humidity environment.

Other operational concerns are described in following chapters:

- Operating and diagnostic procedures, including maintenance and testing: Chapter 16.

4.6 Functional Design of Reactivity Control System

This section demonstrates that the control rod drive system (CRDS) provides the required functional performance and is properly isolated from other equipment. The required functional performance involves the capability to effect a safe shutdown, respond within acceptable limits during anticipated operational occurrences, and prevent or mitigate the consequences of anticipated operational occurrences and postulated accidents.

This section describes how the design of the CRDS conforms to the requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A (Reference 4.6-1): 4, 23, 25, 26, 27, 28, and 29, as well as to the requirements of 10 CFR 50.62. (Reference 4.6-2)

Information in this section is organized into subsections that provide information on the CRDS (Subsection 4.6.1), describe evaluations of this system (Subsection 4.6.2), describe testing and verification of the system (Subsection 4.6.3), provide information on combined performance of reactivity systems (Subsection 4.6.4), and describe evaluations of combined performance of these systems (Subsection 4.6.5). This section refers to other chapters and sections for certain information to avoid duplication.

The reactivity control systems for the US-APWR include both the mechanical reactivity control of the control rods and the chemical reactivity control of the emergency core cooling system (ECCS). No credit for the reactivity control capabilities of the chemical and volume control system (CVCS) is taken for anticipated operational occurrences and postulated accidents described in Chapter 15.

4.6.1 Information for Control Rod Drive System

Subsection 3.9.4 describes the control rod drive mechanism (CRDM) in detail. This section includes drawings of the CRDM, component descriptions and characteristics. The rod control cluster assembly (RCCA) is inserted in the core by gravity, if electrical power of the CRDM coils is cut off. Electrical power to the CRDM coils is cut off by the reactor trip breakers which have under voltage attachment as the fail safe actuation device. Hydraulic systems are not used in the control of the CRDS.

The instrumentation and controls for reactor trip and reactor control are described in Sections 7.2 and 7.7 respectively. The cooling system for the CRDS is described in Subsection 9.4.6.

4.6.2 Evaluations of the CRDS

The CRDS ~~takes a~~is part of the reactor trip system. ~~The d~~Detail of the failure mode and effects analysis (FMEA) on the CRDS is described in Reference 4.6-3. ~~The~~ FMEA of the reactor trip system is ~~also performed as~~ described in Section 7.2. These analyses demonstrate that the CRDS performs a reactor trip when plant parameters exceed~~s~~ the reactor trip setpoint. By this performance, the reactor is placed in a subcritical condition with any assumed credible failure of any single active component, in compliance with GDC 25.

Chapter 5

US-APWR DCD Chapter 5 Rev.2, Tracking Report Rev.7 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-1	5.2.1.1	RAI No.264 Second Amend Question No.05.02.01.01-1, 12/15/2009 Added "Any proposed changes by the COL applicant in the use of the ASME Code editions or addenda specified in the US-APWR DCD will require NRC approval prior to implementation."
5.2-5	Table 5.2.1-2 (Sheet 2 of 2)	RAI No.575 Question 05.02.01.02-7, 5/07/2010 Added Code Case N-782 to the table.
5.2-20	5.2.3.3.2	RAI No.644 Amend Question No.05.02.03-27 (not yet delivered official response) Changed the first sentence of the third paragraph to "Preheating temperatures used during the fabrication of carbon steel and low-alloy steel components follows the recommendations of ASME Code, Section III, Appendix D, Article-1000, with the exception of the first and second passes of circumferential joint welding in low alloy steel, which utilizes a minimum preheat temperature of 122F. Subsequent passes are performed using the recommended preheat temperatures listed in ASME Code, Section III, Appendix D, Article-1000", and add sentence "All welding procedures are qualified at the minimum preheat temperature" after the third sentence of the third paragraph.
5.2-22	5.2.3.4	RAI No.644 Question 05.02.03-28, 11/08/2010 Changed the third sentence of the fourth paragraph to "For cast austenitic stainless steel components used in the RCPB, RPV internals and ESF systems, with service temperatures greater than 482°F, the delta ferrite content is limited to less than or equal to 20% for low molybdenum (0.5 wt% maximum) content statically cast materials, less than or equal to 14% for high molybdenum (2.0-3.0 wt%) content statically cast materials, and less than or equal to 20% for high molybdenum content centrifugally cast materials. Ferrite content will be calculated using Hull's equivalent factors method as described in NUREG/CR-4513, Rev. 1 (May 1994).

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-22 to 23	5.2.3.4.1	RAI No.644 Question 05.02.03-29, 11/08/2010 Added sentence after the second paragraph "During the detailed design of RCPB piping and components, MHI will determine if there are local areas where flow stagnation may be present resulting in dissolved oxygen content greater than 0.10 ppm. For piping and components where the above condition exists, stainless steel with a carbon content less than or equal to 0.03% will be used."
5.2-26 5.2-27 5.2-28 5.2-29	Table 5.2.3-1	RAI No.289 Amend Question No.05.02.03-12, 2/26/2010 Added note "(*)Maximum carbon content will be controlled under 0.05% (heat analysis) and 0.06% (product analysis) when standard grade stainless steel is used" and marked (*) to appropriate locations in the table.
5.2-26 5.2-29	Table 5.2.3-1	RAI No.644 Question 05.02.03-26, 11/08/2010 Added note "(**): Chemical composition for use in the core beltline region will be limited as shown in Section 5.3, Table 5.3-1." and marked (**) to appropriate locations in the table.
5.2-26 5.2-27 5.2-28	Table 5.2.3-1	RAI No.540 Question 05.02.03-19, 6/04/2010, and RAI No.644 Question 05.02.03-26, 11/08/2010 Change the material class of pressure boundary welds (Low alloy steels) of Reactor Vessel, Steam Generator, and Pressurizer.
5.2-35	Subsection 5.2.4.1.2	Correction (editorial correction) Delete "ultrasonic".
5.2-37	Subsection 5.2.5.1	Correct the title of reference to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" [RAI 549-4390, Question 05.02.05-12]
5.2-39	Subsection 5.2.5.4	Correction (editorial correction) In third paragraph, added "change in" before "leakage rate" for clarity.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-40	Subsection 5.2.5.4.1.1	RAI: No. 521, 14.02-120 Changed “A leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators as stated in positions 5 and 7 of regulatory guide 1.45.” to “A change in leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators, consistent with regulatory positions 2.2 and 3.3 of regulatory guide 1.45.”
5.2-41	Subsection 5.2.5.4.1.2 Subsection 5.2.5.4.1.3 Subsection 5.2.5.4.1.4	In third paragraph, added “change in” before “leakage rate” for clarity. In second paragraph, added “change in” before “leakage rate” for clarity. In second paragraph, added “a change in” before “leakage ” for clarity.
5.2-43	Subsection 5.2.5.7	RAI: No. 521, 14.02-120 Added to the beginning of the first paragraph as follows: “Consistent with Regulatory Position C.2.5 of RG 1.45, leakage monitoring systems, including those with location detection capability, have provisions to permit calibration and testing during plant operation, as appropriate.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-43 and 5.2-44	Subsection 5.2.5.8	RAI: No. 521, 14.02-120 Changed "In accordance with the position 9 of regulatory guide 1.45 the limiting condition for identified and unidentified reactor coolant leakages are identified in the Chapter 16. Subsections 3.4.13 addresses RCS leak limits. Subsection 3.4.15 addresses RCS leak detection instrument requirements. The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the Technical Specification (TS) limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value." to "<u>In accordance with the position 4.1 of regulatory guide 1.45, the limiting conditions for identified, unidentified, RCPB and intersystem reactor coolant leakages are identified in the Chapter 16 Technical Specifications (TS). Subsections 3.4.13 and 3.4.14 address RCS operational leakage and pressure isolation valve (intersystem), leak limits, respectively. Subsection 3.4.15 addresses RCS leak detection instrument requirements. The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the TS limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value. In accordance with the guidance in RG 1.45 position C.2.1, the procedure includes the collection of leakage to the containment from unidentified sources so the total flow rate can be detected, monitored and quantified for flow rates greater than or equal to 0.05 gal/min.</u>"
5.2-45	5.2.6 COL 5.2(4) COL 5.2(5)	Correction (editorial correction) Change the phrase from "*The COL applicant addresses and develops the implementation milestone…*" to "*The COL applicant provides and develops the implementation milestone….*"

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-45	5.2.6 COL 5.2(11)	RAI No.264 Second Amend Question No.05.02.01.01-1, 12/15/2009 Change the phrase from "the *ASME Code editions or addenda other than specified in Table 5.2.1-1 or 10 CFR 10 CFR 50.55a is used*" to "*ASME Code editions or addenda other than those specified in Table 5.2.1-1 will be used.* "
5.2-46	Subsection 5.2.7 Reference 5.2-15	Correct the title of reference to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" [RAI 549-4390, Question 05.02.05-12]
5.2-48	5.2.7 Reference 5.2-40	RAI No.644 Question 05.02.03-28, 11/08/2010 Deleted reference 5.2-40.
5.3-16	Subsection 5.3.2.1.5	Correction (editorial correction) Replace subscript of function from "K_{lt}" to "K_{lc}" and "K_{lm}" to "K_{lt}".
5.3-19	Subsection 5.3.3.1	Correction (editorial correction) ・ Add "and" before ""outlet". Delete "and direct vessel injection (DVI)".
5.3-25	Subsection 5.3.3.7	Correction (editorial correction) ・ Delete "clad" Add "inside and" before ""outside.
5.3-29	Subsection 5.3.5	Consistency Reflect latest revision of Technical Report MUAP-09016
5.4-1 5.4-2	5.4.1.1.2	RAI N0.274 Amend Question No.05.04.01.01-3, (not yet delivered official response.) Added the article number of the ASME Code as acceptance criteria.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.4-2	Subsection 5.4.1.1.2, Third paragraph	RAI: No. 274, 05.04.01.01-3 Changed “The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel should be inspected for critical dimensions after the over speed test so that any dimensional changes can be detected. Qualified test procedure and the acceptance criteria should be decided with respect to this test procedure.” to “The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel will be inspected for critical dimensions after the over speed test so that any dimensional changes can be detected. With respect this test procedure, it should be decided qualified test procedure and the acceptance criteria.”
5.4-35	5.4.7.1	Editorial: clarify the language Changed “the” to “an” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
5.4-36	5.4.7.1	Added “and cooldown” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
5.4-37	5.4.7.1	RAI No. 638, Question No. 07.06-24 Added “and design pressure with sufficient wall thickness to withstand the RCS pressure without rupture”
5.4-38	Subsection 5.4.7.2.1	Add “valve” at the end of second paragraph due to editorial clarification.
5.4-42	Subsection 5.4.7.2.3.1, Second paragraph, last sentence	Editorial: clarify the language Changed “Once the pressurizer steam bubble formation is complete, the RHRS would be isolated from the RCS.” to “Once the pressurizer steam bubble formation is complete, the RHRS is isolated from the RCS.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.4-45	5.4.7.2.3.6 C.	RAI No. 601, Question No. 19-437 Replaced “Redundant water” with “Water” Added “Redundant narrow range water level instrument and a mid-range water level instrument, which are shown in Figure 5.1-2 (Sheet 3 of 3), are provided to measure mid-loop water level.” and “A temporary mid-loop water level sensor that measures the RCS water level with reference to pressure at the reactor vessel head vent line and cross over leg is installed in addition to these permanent water level sensors to cope with surge line flooding events.”
5.4-70	Subsection 5.4.10.2.1	Added “At least 120 kW capacity is required for the heaters in the backup groups A, B, C and D each to maintain the RCS pressure near normal operating pressure.” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
5.4-80	Figure 5.4.10-1	Replace figure of pressurizer
5.4-91	Table 5.4.12-1	Replaced “lbm/h” with “lbm/sec”. Editorial

5.2 Integrity of Reactor Coolant Pressure Boundary

This section describes the measures to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) during plant operation. Title 10 CFR 50.2 (Ref.5.2-1) defines the RCPB as those pressure-containing components such as pressure vessels, piping, pumps, and valves which are part of the reactor coolant system (RCS), or connected to the RCS up to, and including, the following:

1. The outermost containment isolation valve in system piping that penetrates the containment.
2. The second of two valves closed during normal plant operation in the system piping that does not penetrate the primary reactor containment.
3. The RCS safety and relief valves.

Components which are part of the RCPB but not discussed in this section are described in the following reference sections:

1. RCPB components which are part of the emergency core cooling system (ECCS) are described in Section 6.3**.**
2. RCPB components which are part of the chemical and volume control system (CVCS) are described in Subsection 9.3.4
3. Design loading, stress limits, and analyses applied to the RCS and ASME Code Class 1, 2, and 3 are provided in Subsection 3.9.1 and Subsection 3.9.3

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50, Section 50.55a

RCPB components are designed and fabricated in accordance with 10 CFR 50.55a (Ref. 5.2-3) which requires compliance with the requirements for Class 1 components in the American Society of Mechanical Engineers (ASME) Code. Some of the components such as the isolation valves and the flow restricting device meet the exclusion requirements of 10 CFR 50.55a(c) (2) (Ref. 5.2-3) and are classified as Quality Group B in accordance with Regulatory Guide 1.26 (Ref. 5.2-6). The quality group classification for the RCPB is in accordance with GDC-1 of 10 CFR 50 Appendix-A (Ref. 5.2-4) and is identified in Subsection 3.2.2, Table 3.2-2. The quality group classification is used to determine the appropriate sections of the ASME Code or other standards to be applied to the components.

The applicable edition and addenda of the ASME Code applied in the design of each Class 1 component are listed in Table 5.2.1-1.The use of Code editions and addenda issued and endorsed by the NRC subsequent to the design certification is permitted if such Code updates are included in a revised 10 CFR 50.55a by the NRC. Any proposed changes by the COL applicant in the use of the ASME Code editions or addenda specified in the US-APWR DCD will require NRC approval prior to implementation. Use of the ASME Code editions or addenda other than specified in Table 5.2.1-1 or 10 CFR 50.55a require NRC approval prior to implementation. Proposed inspections, tests,

Table 5.2.1-2 ASME Code Cases (Sheet 2 of 2)

Code Case Number	Title	Applicable components
N-729-1	Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1	RV
N-782	Use of Code Editions, Addenda, and Cases Section III, Division 1	RCPB Class 1 Components and Piping
OMN-13	Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants	Piping Supports
2142-2	F-Number Grouping for Ni-Cr-Fe Filer Metals Section IX (Applicable to all Sections, including Section III, Division 1, and Section XI)	RV, SG, Pressurizer

Fracture toughness properties of reactor vessel materials are covered in Subsection 5.3.1.5, and fracture toughness of threaded fastener materials is covered in Subsection 5.2.3.6.

Pressure retaining materials comply with Appendix G of 10CFR50 and with Article NB-2300 and Appendix G of Section III of the ASME Code. (Ref.5.2-5, Ref.5.2-22) The RT_{NDT} is established for all required pressure retaining materials used in the Class 1 vessels. The actual as procured fracture toughness data will be submitted to the NRC staff at a predetermined time e.g., ITAAC.

For the ferritic materials used for piping, pumps, and valves of the RCPB, impact testing is performed in accordance with NB-2332 for maximum thickness of 2.5 inches, and in accordance with NB-2331 for thicknesses greater than 2.5 inches.

Calibration of instruments and equipment is performed as required by the ASME Code, Section III, NB-2360.

5.2.3.3.2 Control of Welding

All welding is conducted using procedures in compliance with the rules of ASME Code Sections III and IX to avoid cold cracking and embrittlement of the welded materials. Preheating is performed for Class 1 component weld joints in accordance with the qualified welding procedure specifications (WPS) in compliance with ASME Code Sections III and IX.

Minimum preheat and maximum interpass temperatures for welding of low alloy steel components of the RCPB are as follows.

~~In accordance with RG 1.50 "Control of Preheat Temperature for Welding Low Alloy Steel" (Ref. 5.2-38), minimum preheat temperatures are generally kept higher than those listed in Appendix D of ASME Section III, which is 250°F for P-No.3 material and 50°F for P-No.1 material. However, minimum preheat temperatures below this temperature may be applied to the first passes of P-No.3 material provided a welding procedure qualification demonstrates acceptable weld integrity and quality.~~ Preheating temperatures used during the fabrication of carbon steel and low-alloy steel components follows the recommendations of ASME Code, Section III, Appendix D, Article-1000, with the exception of the first and second passes of circumferential joint welding in low alloy steel, which utilizes a minimum preheat temperature of 122F. Subsequent passes are performed using the recommended preheat temperatures listed in ASME Code, Section III, Appendix D, Article-1000. Taking into consideration the weldability and quality of the product, preheating above 122°F is applied to the first and second passes of circumferential joint welding. Preheating for more than 250°F is applied to subsequent passes in accordance with Appendix D of ASME Section III. All welding procedures are qualified at the minimum preheat temperature. Maximum interpass temperatures for production welding shall be those specified in welding procedure qualification. For low alloy steels and carbon steels, generally the maximum interpass temperature is 500°F for both materials.

Hydrogen is removed by either post heating at a temperature and time sufficient to preclude the effects of hydrogen assisted cracking, or by maintaining preheat until post weld heat treatment is performed. Post-weld baking is maintained at a temperature of of the reactor coolant are well controlled to prevent SCC of austenitic stainless steel. Field experience has shown that de-oxygenated, hydrogenated PWR primary water does not cause SCC in sensitized materials, unlike BWRs with oxygenated water (Ref. 5.2-31).

Although the water chemistry condition is well controlled, good metallurgical and environmental practices will be followed to further assurance of avoiding SCC, such as:

- Process control of cleaning and protection against contamination
- Use of materials in the final heat treated condition
- Control of welding processes and procedures to avoid heat-affected zone (HAZ) sensitization

Subsections 5.2.3.4.1 and 5.2.3.4.2 address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and present the methods and controls to avoid sensitization and to prevent intergranular attack (IGA) of austenitic stainless steel components.

Cast austenitic stainless steel components used in light-water reactors can be susceptible to thermal aging embrittlement due to the formation of a Cr-rich phase from the decomposition of the ferrite phase by exposing the material to elevated temperatures. Cast austenitic stainless steels that are used in US-APWR RCPB components are categorized based on molybdenum content, casting method, and δ-ferrite level, and examined based on ASME Section XI requirements. (Ref. 5.2-39) ~~Hull's equivalent factor method or Schoefer Diagram in ASTM A800 "Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof" is used to calculate δ-ferrite level. (Ref. 5.2-40)~~ For cast austenitic stainless steel components used in the RCPB, RPV internals and ESF systems, with service temperatures greater than 482°F, the delta ferrite content is limited to less than or equal to 20% for low molybdenum (0.5 wt% maximum) content statically cast materials, less than or equal to 14% for high molybdenum (2.0-3.0 wt%) content statically cast materials, and less than or equal to 20% for high molybdenum content centrifugally cast materials. Ferrite content will be calculated using Hull's equivalent factors method as described in NUREG/CR-4513, Rev. 1 (May 1994).

5.2.3.4.1 Cleaning and Contamination Protection

Control of cleaning and protection against contamination is necessary to prevent SCC of austenitic stainless steel.

Austenitic stainless steels are susceptible to SCC under conditions where dissolved oxygen and halogens (such as chloride and fluoride) are present. The RCS water chemistry is controlled to prevent the intrusion of these aggressive substances. During normal operation, chloride is kept below 0.15 ppm and dissolved oxygen below 0.1 ppm (Table 5.2.3-2, US-APWR water chemistry limits). The effectiveness of these measures has been demonstrated by both laboratory tests and PWR operating experience. Therefore, the possibility of SCC of austenitic stainless steel is considered minimal in the US-APWR plant.

During the detailed design of RCPB piping and components, MHI will determine if there are local areas where flow stagnation may be present resulting in dissolved oxygen content greater than 0.10 ppm. For piping and components where the above condition exists, stainless steel with a carbon content less than or equal to 0.03% will be used.

MHI imposes strict process controls for cleaning and protection against contamination for austenitic stainless steel materials during all stages of component manufacture. These process controls are applied to RCPB components, components of systems that are required for reactor shutdown, systems required for emergency core cooling, and reactor vessel internals. Exposure to contaminants is avoided by carefully controlling all cleaning and processing materials that contact the stainless steel during manufacture and installation. Halogens, especially chloride and fluoride, and their compounds are controlled in the expendable materials, such as packing materials, gaskets, insulation, tape, and lubricants.

Water quality of cleaning solutions, hydrostatic test solutions, and wet layup solutions are also controlled. The water used for final cleaning or flushing of RCPB Class 1 components is demineralized water with chloride and fluoride concentrations less than 0.15 ppm. MHI standards also provide for control of tools used in abrasive work operations such as grinding. Non-sensitization of the material is verified using the methods discussed in Regulatory Guide 1.44 ”Control of the use of sensitized stainless steel.” Pickling of sensitized stainless steel is prevented. Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or SCC.

Adhesive tapes and binding agents on the surfaces of the austenitic stainless steel are completely removed using suitable solvent.

Low melting point materials such as lead, zinc, tin, antimony, cadmium, indium, mercury, and their compounds can have a detrimental effect on austenitic stainless steel and nickel-chromium-iron alloys, therefore low melting point materials are strictly controlled during the manufacturing and installation processes.

Threaded fastener lubricants containing molybdenum sulphide are prohibited from being used.

The surface cleanliness achieved by the MHI procedures satisfies the requirements of Regulatory Guides 1.37 (Ref.5.2-12) and 1.44.

5.2.3.4.2 Solution Heat Treatment Requirements

Sensitized stainless steel is more susceptible to SCC than non-sensitized stainless steel. Solution heat treated austenitic stainless steel exhibits good resistance against SCC. The types of austenitic stainless steels listed in Table 5.2.3-1 are solution heat treated as required by ASME Section II.

SCC of austenitic stainless steel, especially at the location of the HAZ, has been observed in BWR plants. Intergranular stress corrosion cracking of weld-sensitized austenitic stainless steels has since been extensively investigated (Ref. 5.2-30). The

Table 5.2.3-1 Reactor Coolant Pressure Boundary Material Specifications (Sheet 1 of 4)

Component	Material	Class, Grade, or Type
Reactor Vessel Parts		
Top head dome	SA-508	Gr. 3 Cl.1
Shell and flange ring forgings	SA-508	Gr. 3 Cl.1(**)
Bottom head dome	SA-508	Gr. 3 Cl.1
Nozzle forgings	SA-508	Gr. 3 Cl.1
Nozzle safe ends	SA-182	Gr. F316(*) or F316LN
Upper head penetration nozzles	SB-167	UNS N06690 (Thermally Treated 690)
Vent pipe	SB-167	UNS N06690 (Thermally Treated 690)
Radial support	SB-166	UNS N06690 (Thermally Treated 690)
Reactor vessel closure stud bolts, nuts, washers	SA-540	Gr.B24
Cladding, buttering and welds	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L ER316L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
	Type 308L/309L Stainless Steel Strip Electrode	-
Pressure boundary welds (Low alloy steel)	SFA-5.5	E9016-G(**)
	SFA-5.23	F9P4-EG-G, ~~E9P4-EGN-GN~~ F9P4-EGN-GN(**)
	SFA-5.28	ER80S-G
Pressure boundary welds (Stainless steel or Ni-base alloy)	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
Steam Generator Parts		
Pressure forgings (including nozzles)	SA-508	Gr. 3 Cl.2
Pressure plates	SA-533	Type B Cl.2
Tubes	SB-163	UNS N06690 (Thermally Treated 690)

Table 5.2.3-1 Reactor Coolant Pressure Boundary Material Specifications (Sheet 2 of 4)

Component	Material	Class, Grade, or Type
Nozzle safe ends	SA-182	Gr. F316(*) or F316LN
Closure Stud bolts	SA-193	Gr. B7
Closure nuts	SA-194	Gr. 4
Cladding, buttering and welds	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
	Type 308L/309L Stainless Steel Strip Electrode	-
Pressure boundary welds (Low alloy steel)	SFA-5.5	E9016-G E10016-G
	SFA-5.23	F9P4-EG-G ~~E9P4-EGN-GN~~ ~~P10P2-EG-G~~ F10P2-EG-G
	SFA-5.28	ER80S-G ER90S-G
Pressure boundary welds (Stainless steel or Ni-base alloy)	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
Pressurizer Parts		
Pressure forgings	SA-508	Gr. 3 Cl.1 or Cl.2
Pressure plates	SA-533	Type B Cl.1 or 2
Nozzle safe ends	SA-182	Gr. F316(*) or F316LN
Heater sleeves	SA-182	Gr. F316(*) or F316LN
	or SB 167	UNS 06690
Closure Stud bolts	SA-193	Gr. B7
Closure nuts	SA-194	Gr. 4

Table 5.2.3-1 Reactor Coolant Pressure Boundary Material Specifications (Sheet 3 of 4)

Component	Material	Class, Grade, or Type
Cladding, buttering and welds	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
	Type 308L/309L Stainless Steel Strip Electrode	-
Pressure boundary welds (Low alloy steel)	SFA-5.5	E9016-G E10016-G
	SFA-5.23	F9P4-EG-G ~~E9P4-EGN-GN~~ ~~P10P2-EG-G~~ F10P2-EG-G
	SFA-5.28	ER80S-G ER90S-G
Pressure boundary welds (Stainless steel or Ni-base alloy)	SFA-5.4	E309L-16 E308L-16
	SFA-5.9	ER309L ER308L
	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7
	Code Case 2142-2 UNS N06054	-
Reactor Coolant Pump Parts		
Pressure casting	SA-351	Gr. CF8(*)
Pressure forgings	SA-182	Gr. F304(*) or F304LN Gr.F316(*) or F316LN
Tubes and pipes	SA-213 SA-312	TP 316(*) TP 316(*)
Flywheel	SA-533	Type B Class 1
Closure Stud bolts, Nuts, Washer	SA-540	Gr. B24 Cl.4 and Cl.2
Reactor Coolant Piping		
Main coolant pipe and elbow	SA-182 or SA-336	Gr. F316(*) or F316LN
Main coolant branch nozzles	SA-182	Gr. F316(*) or F316LN
Pressure Boundary Welds	SFA-5.4 SFA-5.9	E316L-16 ER316L
Surge line, spray line, and other RCS piping	SA-312 or SA-376	TP 316(*), or 316LN or 316L

Table 5.2.3-1 Reactor Coolant Pressure Boundary Material Specifications (Sheet 4 of 4)

Component	Material	Class, Grade, or Type
Auxiliary Pressure Vessels, Tanks		
Pressure Plates	SA-240	Type 304(*) or Type304L
Pressure Forgings	SA-182	Gr. F304(*) or Type 304L
	SA-105	-
Valves		
Bodies	SA-351	CF3A, CF3M, CF8(*), CF8M(*)
	SA-182	Gr.F304(*), F304L, F304LN Gr.F316(*), F316L, F316LN
Bonnets	SA-351	CF3A, CF3M, CF8(*), CF8M(*)
	SA-240	Type 304(*), 304L, 304LN Type 316(*), 316L, 316LN
	SA-182	Gr.F304(*), F304L, F304LN Gr.F316(*), F316L, F316LN
Disks	SA-564	Type 630
	SA-479	Type 304(*), 304L, 304LN Type 316(*), 316L, 316LN
	SA-351	CF3A, CF3M, CF8(*), CF8M(*)
	SA-182	Gr.F304(*), F304L, F304LN Gr.F316(*), F316L, F316LN
	SB-637	UNS N07718
Stems	SA-564	Type 630
	SA-479	Type 304(*), 304L, 304LN Type 316(*), 316L, 316LN Type 403
	SB-637	UNS N07718
Closure Stud Bolts	SA-453	Gr.660
	SA-193	Gr.B7, B16
	SA-564	Type 630
Closure Nuts	SA-453	Gr.660
	SA-193	Gr.B7, B16
	SA-194	Gr.6 or 8

(Note) Material specifications for Reactor Vessel Internals and Control Rod Drive Mechanism are described at Section 4.4.
(*) Maximum carbon content will be controlled under 0.05% (heat analysis) and 0.06% (product analysis) when standard grade stainless steel is used
(**): Chemical composition for use in the core beltline region will be limited as shown in Section 5.3, Table 5.3-1.

The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Subarticle IWA-2200, IWB-2000 and Table IWB-2500-1 of the ASME Code, Section XI (Ref. 5.2-25). Qualification of the ultrasonic inspection equipment, personnel, and procedures is in compliance with Appendix VII and Appendix VIII of the ASME Code, Section XI (Ref. 5.2-25). The liquid penetrant method, eddy current, ~~ultrasonic~~ or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations.

Personnel performing nondestructive examinations will be qualified and certified using a written practice in accordance with ASME Code Section XI, Article IWA-2300, "Qualification of Nondestructive Examination Personnel," as modified by 10 CFR 50.55a(b)(2)(xviii). The methods, procedures, and requirements for ultrasonic examination of RV welds are in accordance with the requirements of Appendix VIII of Section XI of the ASME Code.

Performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect flaws is in accordance with the requirements of Appendix VIII of Section XI of the ASME Code.

Sufficient clearance is provided around pipe or component welds requiring volumetric or surface examination for ISI.

Inspections carried out remotely are considered for high radiation areas to support as low as reasonably achievable (ALARA) goals. Remote inspections are also considered in areas where physical limitations restrict or prevent manual methods.

5.2.4.1.3 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWA-2400 and IWB-2400 of the ASME Code, Section XI (Ref. 5.2-25). The interval may be reduced or extended by as much as one year so that inspections are concurrent with plant outages. It is intended that inservice examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.1.4 Evaluation of Examination Results

Examination results are evaluated according to ASME Section XI, IWB-3000, with flaw indications according to IWB-3400 and Table IWB-3410-1. Repair and replacement activities, if required, are according to IWA-4000 of the ASME Code, Section XI (Ref. 5.2-25).

5.2.4.1.5 System Pressure Tests

System pressure tests comply with IWB-5000 of the ASME Code, Section XI and the technical specification requirements for operating limitations during heat-up, cool-down, and system hydrostatic pressure testing. These system pressure tests are included in the design transients defined in Subsection 3.9.1. This subsection discusses the transients included in the evaluation of fatigue of Class 1 components due to cyclic loads.

5.2.5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The reactor coolant pressure boundary (RCPB) leak monitoring system provides a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. This system provides information which permits the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.

5.2.5.1 Design Bases

The leak monitoring system is designed in accordance with the requirements of General Design Criterion 30 and the regulatory guidance as identified below:

- General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" of Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," to provide a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage.
- Regulatory Guide 1.45, "~~Reactor Coolant Pressure Boundary Leakage Detection Systems~~Guidance on Monitoring and Responding to Reactor Coolant System Leakage" (Ref.5.2-15).
- Regulatory Guide 1.29, "Seismic Design Classification" (Ref.5.2-7).

5.2.5.2 Classification of Leakage

RCPB leakage is classified as either identified or unidentified leakage in accordance with the guidance of position 1 of regulatory guide 1.45.

Identified leakage includes the following:

- Leakage into closed systems such as pump seals or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank.
- Leakage into the containment atmosphere for which the location is identified, without interfering with the unidentified leakage detection system or is identified as leakage from other than the RCPB.
- Leakage into auxiliary systems and secondary systems.

Unidentified leakage is all other leakages.

5.2.5.3 Detection of Identified Leakage

Identified leakage other than intersystem leakage, such as pump seal or valve packing, is directed to the C/V reactor coolant drain tank where it is monitored by tank pressure, temperature, and level indications.

An important identified leakage path for reactor coolant into other systems is a flow to the secondary side of the steam generator (SG) through the SG tubes.

increase the piping temperature. A surface mounted RTD which is located downstream these valves is installed on the bottom of each target pipe. Leakage past these valves is detected by these RTDs and alarms in the MCR.

E. RHR Emergency Letdown Lines

The RHR emergency letdown lines are isolated from the RCS by normally closed motor operated valves SIS-MOV-031A and -031D and SIS-MOV-032A and -032D. Leakage past these valves will increase the piping temperature. A surface mounted RTD which is located downstream these valves is installed on the bottom of each target pipe. Leakage past these valves is detected by these RTDs and alarms in the MCR.

F. Reactor Head Seal

Seal leakage is detected by means of two monitoring tubes in the upper shell flange, one located between the inner and outer O-rings, and one located outside the outer O-rings. Piping and associated valves direct any leakage to the C/V reactor coolant drain tank.

A surface mounted RTD, installed on the bottom of the common pipe, sends a high temperature alarm signal to the MCR indicating the possibility of a leakage from the Reactor Vessel head seal.

G. Component Cooling Water System

Leakage from the RCS to the component cooling water (CCW) system is detected by the CCW radiation monitors and/or increase in the CCW surge tank level.

5.2.5.4 Detection of Unidentified Leakage

Indications of unidentified coolant leakage into the containment are provided by an air particulate radioactivity monitor, an airborne gaseous radioactivity monitor, an air cooler condensate flow rate monitoring system, and a containment sump level and flow monitoring system.

In normal operation, these monitors show a background level that is indicative of the normal level of unidentified leakage inside the containment. Variations in airborne radioactivity or specific humidity above the normal level signify an increase in unidentified leakage rates and signal to the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage.

The sensitivity and response time of leakage detection equipment for unidentified leakage is such that a change in leakage rate, or its equivalent, of 0.5 gpm can be detected in less than an hour.

The methods employed for detecting leakage to the containment from unidentified sources are:

- Containment sump level

- Containment airborne particulate radioactivity
- Containment airborne gaseous radioactivity
- Condensate flow rate from air coolers.

Additionally, humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment. They do not quantify the reactor coolant leakage.

5.2.5.4.1 System Description of Unidentified Leakage detection

5.2.5.4.1.1 Containment Sump Level and Flow Monitoring System

Any leakage inside the containment from the RCPB and other components, not otherwise identified, condenses and flows by gravity through the floor drains and other drains to the containment sump, where the sump level meter measures the increase in the sump level indicating the leak rates. Indication of increasing sump level is transmitted from the sump to the MCR by means of a sump level transmitter and recorded.

A change in leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators, consistent with ~~as stated in~~ regulatory positions ~~5~~2.2 and ~~7~~3.3 of regulatory guide 1.45.

The sump level monitoring system is qualified for a safe shutdown earthquake.

5.2.5.4.1.2 Containment Airborne Particulate Radioactivity Monitor

In US-APWR, this monitor corresponds to the containment radiation monitor (RMS-RE-040). Refer to Chapter 11, Subsection 11.5.2. The containment airborne particulate radioactivity monitor performs continuous sampling of the containment air and measures the radiation level in the particulate. This monitor is qualified for a safe-shutdown earthquake (SSE). An air sample is drawn outside the containment and passed through a gamma monitor that monitors its gamma rays in radioactive particulate. After passing through the monitor, the sample is returned via the closed system to the containment atmosphere. The measuring range for the monitor is from 1×10^{-10} μCi /cm^3. An indication of the monitor counting rate is provided to the MCR and electronically recorded.

The detection sensitivity of the airborne particulate radioactivity monitor for reactor coolant leak rate depends on conditions, such as radioactive concentration in the reactor coolant and a distribution coefficient of radioactive particles to the containment atmosphere.

In addition, provided that a radioactive concentration of airborne particulate in the containment is within the measuring range of the airborne particulate radioactivity monitor, an alarm is adjustable to actuate upon detection of a severalfold increase.

Assuming that corrosion and activation product concentration in the reactor coolant is 2×10^{-1} μCi /g (Na-24,Cr-51,Zn-65,Mn-54,56, Co-58,60, Fe-55,59) and the distribution coefficient is 0.3, after leak occurrence, a change in leak rate of 0.5 gpm can be detected within one hour.

5.2.5.4.1.3 Containment Airborne Gaseous Radioactivity Monitor

In US-APWR, this monitor corresponds to the containment radiation monitor (RMS-RE-041). Refer to Chapter 11, Subsection 11.5.2. The containment airborne gaseous radioactivity monitor measures the radiation level in the gas stream from the containment atmosphere. This monitor is equipped with the scintillation monitor, which performs continuous sampling taken from the air inside the containment to outside the containment and continuously measures the sample gas after passing through the containment airborne particulate radioactivity monitor. The measured gas returns to inside the containment. The sensitivity of the monitor is 5 $\times 10^{-7}$ μCi /cm^3. The counting rate of the monitor is provided to the MCR and recorded.

Assuming that Xe-133 concentration in the reactor coolant is 3.2 μCi /g, after leak occurrence, a change in leak rate of 1 gpm can be detected within one hour.

This monitor is qualified for seismic events not requiring a plant shutdown.

5.2.5.4.1.4 Containment Air Cooler Condensate Flow Rate Monitoring System

The containment air cooler condensate flow rate monitoring system consists of a containment cooler drain collection header, a vertical standpipe, valves, and standpipe level instrumentation. The monitoring system collects the condensate from the cooling coils of the containment recirculation unit coolers and CRDM cooling unit and enables volume measurements. Both humidity in the containment and the collected condensate which start to increase are associated with indication of leakage. Under equilibrium state, the fluid volume, which condenses in HVAC units inside the containment, is equivalent to evaporated primary coolant volume at the leak source. The condensation from the containment air coolers flows via the collection header to the vertical standpipe. A differential pressure transmitter provides standpipe level signals. The system provides measurements of low leakages by monitoring standpipe level increase versus time. The condensate flow rate is recorded and high alarms are provided in the MCR.

The humidity at the inlet of the HVAC unit cooling coil inside the containment starts to increase by vapor generated from the leak source resulting in the condensate volume increase. During normal operation, a change in leakage of 0.5 gpm can be detected within one hour of detector response time since containment recirculation fans sufficiently circulate the air inside the containment.

This monitoring system is qualified for seismic events not requiring a plant shutdown.

5.2.5.4.2 Additional Unidentified Leakage Detection Methods

A. Charging Pump Operation

C. Containment air cooler condensate flow rate monitoring system - standpipe level

D. Containment sump level and flow monitoring system – sump level

E. Gross leakage detection methods - charging flow rate, letdown flow rate, pressurizer level, VCT level and reactor coolant temperatures are available as inputs for detection by RCS inventory balance. Containment sump levels and pump operation are also available. Total makeup water flow is available from the plant computer for liquid inventory.

F. Containment temperature, pressure, and humidity will only have readouts in the MCR and alarms to indicate occurrence of leakage within the containment. This method is used only to detect leaks and is not used to quantity leak rates.

5.2.5.7 Testing, Calibration and Inspection Requirements

Consistent with Regulatory Position C.2.5 of RG 1.45, leakage monitoring systems, including those with location detection capability, have provisions to permit calibration and testing during plant operation, as appropriate. Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detection equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks. A description of testing and calibration for the containment radioactivity monitoring system is presented in Subsection 11.5.2.

Periodic inspection of the floor drainage system to the containment sump is conducted to check for blockage and ensure unobstructed pathways.

The containment humidity monitoring systems and the containment air cooler condensate flow rate monitoring system are also periodically tested to ensure proper operation and verify sensitivity.

In service inspection criteria, equipment used, procedures, frequency of testing, inspection, surveillance, and examination of the structural and leak-tight integrity of RCPB components are described in Subsection 5.2.4.

5.2.5.8 Limits for Reactor Coolant Leakage Rates within the RCPB

In accordance with the position ~~9~~4.1 of regulatory guide 1.45, the limiting conditions for identified, ~~and~~ unidentified, RCPB and intersystem reactor coolant leakages are identified in the Chapter 16 Technical Specifications (TS). Subsections 3.4.13 and 3.4.14 address~~es~~ RCS operational leakage and pressure isolation valve (intersystem), leak limits, respectively. Subsection 3.4.15 addresses RCS leak detection instrument requirements.
The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the ~~Technical Specification (~~TS~~)~~ limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value. In accordance with the guidance in RG 1.45 position C.2.1, the procedure includes the collection of leakage to the

containment from unidentified sources so the total flow rate can be detected, monitored and quantified for flow rates greater than or equal to 0.05 gal/min.

5.2.6 Combined License Information

COL 5.2(1) ASME Code Cases that are approved in Regulatory Guide 1.84

The COL applicant addresses the addition of ASME Code Cases that are approved in Regulatory Guide 1.84.

COL 5.2(2) ASME Code Cases that are approved in Regulatory Guide 1.147

The COL applicant addresses Code Cases invoked in connection with the inservice inspection program that are in compliance with Regulatory Guide 1.147.

COL 5.2(3) ASME Code Cases that are approved in Regulatory Guide 1.192

The COL applicant addresses Code cases invoked in connection with the operation and maintenance that are in compliance with Regulatory Guide 1.192.

COL 5.2(4) Inservice inspection and testing program for the RCPB

The COL applicant provides ~~addresses~~ and develops the implementation milestone of the inservice inspection and testing program for the RCPB, in accordance with Section XI of the ASME Code and 10 CFR 50.55a.

COL 5.2(5) Preservice inspection and testing program for the RCPB

The COL applicant provides ~~addresses~~ and develops the implementation milestone of the preservice inspection and testing program for the RCPB in accordance with Article NB-5280 of Section III, Division I of the ASME Code.

COL 5.2(6) Deleted

COL 5.2(7) Deleted

COL 5.2(8) Deleted

COL 5.2(9) Deleted

COL 5.2(10) Deleted

COL 5.2(11) ASME Code Edition and Addenda

The COL applicant addresses whether ~~the~~ ASME Code editions or addenda other than specified in Table 5.2.1-1 ~~or 10 CFR 10 CFR 50.55a is~~ will be used.

5.2.7 References

5.2-1 Definitions, NRC Regulations Title 10, Code of Federal Regulations, 50.2.

5.2-2 Contents of Construction Permit and Operating License Applications; Technical Information, NRC Regulations Title 10, Code of Federal Regulations, 50.34.

5.2-3 Domestic Licensing of Production and Utilization Facilities, NRC Regulations Title 10, Code of Federal Regulations, 50.55a.

5.2-4 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 50 Appendix A.

5.2-5 Fracture Toughness Requirements, NRC Regulations Title 10, Code of Federal Regulations, 50 Appendix G.

5.2-6 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Regulatory Guide 1.26, Rev. 4, March 2007.

5.2-7 Seismic Design Classification, Regulatory Guide 1.29, Rev.4, March 2007.

5.2-8 Control of Ferrite Content in Stainless Steel Weld Metal, Regulatory Guide 1.31, Rev. 3, April 1978.

5.2-9 Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Regulatory Guide 1.84, Rev. 34, October 2007.

5.2-10 Control of Electroslag Weld Properties, Regulatory Guide 1.34, Rev.0, December 1972.

5.2-11 Nonmentallic Thermal Insulation for Austenitic Stainless Steel, Regulatory Guide 1.36, Rev.0, February 1973.

5.2-12 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, Regulatory Guide 1.37, Rev. 1, March 2007.

5.2-13 Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components, Regulatory Guide 1.43, Rev.0, May 1973.

5.2-14 Control of the Use of Sensitized Stainless Steel, Regulatory Guide 1.44, Rev.0, May 1973.

5.2-15 ~~Reactor Coolant Pressure Boundary Leakage Detection Systems~~Guidance on Monitoring and Responding to Reactor Coolant System Leakage, Regulatory Guide 1.45, Rev.1, May 2008.

5.2-16 Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, Rev. 0, October 1973.

5.2-34 Contents of Applications; Technical Information, NRC Regulations Title 10, Code of Federal Regulations, 50.47.

5.2-35 Operation and Maintenance Code Case Acceptability, ASME OM Code.

5.2-36 US-APWR Sump Strainer Performance, MUAP-08001, Revision 2, December 2008.

5.2-37 Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, Generic Letter 88-05, March 17. 1998.

5.2-38 Control of Preheat Temperature for Welding of Low-Alloy Steel, Regulatory Guide 1.50, Rev.0, May 1973.

5.2-39 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components, License Renewal Issue No. 98-0030, May 19, 2000.

5.2-40 ~~Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof, ASTM A 800/A 800M-01, 2006.~~ Deleted

5.3.2.1.4 Determination of K_{Im}

K_{Im} is determined by the following equation in accordance with ASME Code Section XI Appendix G.

$$K_{Im} = M_m \times \left(\frac{PR_i}{t_{th}}\right) \qquad \text{(psi)}$$

For inside (1/4-T) axial surface flaws, M_m is determined as follows:

M_m = 1.85 for $\sqrt{t_{th}} < 2$

M_m = $0.926\sqrt{t_{th}}$ for $2 \le \sqrt{t_{th}} \le 3.464$

M_m = 3.21 for $\sqrt{t_{th}} > 3.464$

For outside (3/4-T) axial surface flaws, M_m is determined as follows:

M_m = 1.77 for $\sqrt{t_{th}} < 2$

M_m = $0.893\sqrt{t_{th}}$ for $2 \le \sqrt{t_{th}} \le 3.464$

M_m = 3.09 for $\sqrt{t_{th}} > 3.464$

where,

P = Internal pressure (ksi)

5.3.2.1.5 Determination of P-T Limit Curves

Based on the relationships in Subsections 5.3.2.1.1 to 5.3.2.1.4, relationships can be established for the P-T limit curves. The relationship for heatup and cooldown operations is as follows.

$$P = \frac{K_{Ic} - K_{It}}{2M_m\left(\frac{R_i}{R_o - R_i}\right)}$$

For heatup operation, the inner surface temperatures are higher than the outer surface temperatures, therefore stresses are in compression on the inner surfaces and in tension on the outer surfaces. However, since degradation due to irradiation is higher for the inner surfaces, the above equation is evaluated for both inner (1/4-T) and outer (3/4-T) surfaces, and the limiting pressures determine the limits of operation. For cooldown, only the inner (1/4-T) surfaces are evaluated as tension stresses occur on the inner surfaces, and effects of irradiation are also more conservative for the inner surfaces.

5.3.3.1 Design

Reactor Vessel

The reactor vessel is a pressure boundary component, designed and fabricated in accordance with ASME Code Section III requirements for Class 1 components, whose function is to support and enclose the reactor core internals. With the reactor core internals, the reactor vessel guides the flow of reactor coolant, and also maintains a volume of coolant around the core.

The reactor vessel is a vertical cylindrical pressure vessel, consisting of a vessel flange and upper shell, lower shell, transition ring, hemispherical bottom head and removable upper closure head.

The reactor vessel is fabricated by welding the vessel flange and upper shell, lower shell, transition ring and bottom head. The inlet and outlet ~~and direct vessel injection (DVI)~~ nozzles are welded to the upper shell. The main dimensions of the reactor vessel are shown in figures 5.3-4 and 5.3-5.

The closure head consists of a bolting flange and hemispherical top head dome. The top head dome has penetrations for the control rod drive mechanisms, in-core instrumentation systems and head vent.

Lifting lugs and support lugs for the integrated head package are welded to the outside of the closure head.

The length of the entire reactor vessel, including closure head, is approximately 44.4 feet. The inner diameter at the beltline region is 202.8 inches. The total weight of the reactor vessel (including closure head and stud bolts, but excluding control rod drive mechanisms) is approximately 640 tons.

Wetted surfaces during operation and refueling are clad with stainless steel weld overlay of nominal thickness of 0.2 inches. This includes the vessel shell flange top surface but does not include inside the stud bolt holes.

The design pressure and temperature for the US-APWR reactor vessel is 2,485 psig and 650°F. The design life is 60 years.

As an additional safety precaution, no penetrations are located below the top of the reactor core. This minimizes the potential for a loss of coolant accident by leakage from the reactor vessel, allowing the reactor core to be uncovered. The reactor core is also positioned inside the reactor vessel to limit reflood time in case of an accident. Radial support for the reactor internals is provided by key and keyway joints, located at the lower end of the reactor internals. Radial supports are located on the inner diameter of the reactor vessel and are equally spaced circumferentially.

The interface between the reactor vessel and reactor internals is such that the reactor coolant enters through the inlet nozzle, flows down through the annulus between the reactor vessel and reactor internals, and then flows upward through the core. In addition, achieved using remote, mobile inspection devices. The head insulation is offset from the top head surface by approximately 4 inches to allow access for these inspection devices.

The knuckle transition area of the closure head has relatively high operating stresses within the closure head and is accessible from the outer surface for visual inspection, surface NDE by liquid penetrant or magnetic particle methods, and ultrasonic examination.

The closure head stud bolts, nuts and washers can be inspected periodically using visual, magnetic particle and/or ultrasonic examinations. The stud bolts, nuts, and washers can be removed to dry storage areas during refueling for such inspections.

The upper shell cladding is accessible for inspection during refueling in certain areas above the inlet and outlet nozzles. If necessary, the lower core internals can be removed so that the entire vessel inner surfaces are accessible.

In addition, full-penetration welds in the following areas of the installed reactor vessel are also accessible for NDE:

- Closure head, from the inside clad and outside surfaces. A stand is provided on the operating deck for the closure head to be stored during refueling to facilitate inspections.
- Vessel shell, vessel nozzles, transition ring and bottom head, from the inside clad surface. Completely removable reactor internals facilitate such inspections. Storage space for the reactor internals is provided.
- Welds between the inlet/outlet nozzles and nozzle safe ends, from the inside ~~clad~~ and outside surfaces.
- Welds between the DVI nozzles and nozzle safe ends, from the inside and outside surfaces.
- Field welds between the reactor vessel nozzle safe ends and the main coolant piping, from the outside surfaces. The insulation covering these welds is removable to provide access for the inspections.

Due to factors such as radiation levels and underwater accessibility, some regions of the reactor vessel, particularly those from the vessel inner clad surfaces, may be difficult to access for inservice inspections required by the ASME Code. Several design and manufacturing considerations are made to enhance the inservice inspection results for these regions.

- During manufacturing of the reactor vessel, ultrasonic examinations are performed on internally clad surfaces to an acceptance and repair standard which confirms the cladding bond is sufficient to permit ultrasonic examination of the base metal from the inside surface. These examinations are carried out in accordance with the procedures of ASME Code Section V.
- The inside surfaces of the reactor vessel shells are cylindrical surfaces free of obstructions that may interfere with the inspection equipment.

5.3-21 Nondestructive Examination, ASME Boiler and Pressure Vessel Code, Section V, American Society of Mechanical Engineers, 2001 Edition with 2003 Addenda.

5.3-22 Welding and Blazing Qualifications, ASME Boiler and Pressure Vessel Code Section IX, American Society of Mechanical Engineers, Latest Edition and Addenda.

5.3-23 Rules for In-service Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code Section XI, American Society of Mechanical Engineers, 2001 Edition with 2003 Addenda.

5.3-24 Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, ASTM E-185-82.

5.3-25 Standard Test Method for Linear-Elastic Plane-Strain Fracture Toughness K_{Ic} of Metallic Materials, ASTM E-399.

5.3-26 Standard Test Method for Measurement of Fracture Toughness, ASTM E-1820.

5.3-27 Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

5.3-28 US-APWR Reactor Vessel Pressure and Temperature Limits Report, MUAP-09016 Rev.1 ~~0~~, January 2010 ~~June 2009~~.

5.4 Reactor Coolant System Component and Subsystem Design

This section provides information regarding the performance requirements and design features of the reactor coolant pump, steam generator, reactor coolant piping, main steam flow restrictor, residual heat removal system, pressurizer and discharge system, pressurizer relief tank, and RCS high point vent system.

Information about the reactor coolant system support is described in Subsection 3.8.3.1. Information about the pressurizer safety valve is described in Subsection 5.2.2, and information about the safety depressurization valve (SDV) and depressurization valve (DV) is described in Subsection 5.4.12.

5.4.1 Reactor Coolant Pumps

5.4.1.1 Pump Flywheel Integrity

Integrity of the reactor coolant pump (RCP) flywheel is ensured on the basis of the following design and quality assurance procedures.

5.4.1.1.1 Design Basis

The RCP flywheel is designed, manufactured, and inspected to minimize the possibility of generating high-energy fragments (missiles) under any anticipated operating or accident condition consistent with the intent of the guidelines set forth in Standard Review Plan (SRP) 5.4.1.1 and Regulatory Guide (RG) 1.14 (Ref. 5.4-9).

Calculated stress at operating speed is based on stress due to centrifugal force. Conservatively, 125% of operating speed is selected as the design speed for the RCPs. Flywheels are tested at 125% of the maximum synchronous speed of the motor.

An analysis is performed to predict the critical speed, which is determined the flywheel failure mode of ductile failure, nonductile failure, and excessive deformation of the flywheel. The flywheel is designed that the normal speed is less than the one-half of the lowest of the critical speeds. And it is confirmed that the lowest critical speed is greater than the predicted loss-of-coolant accident (LOCA) overspeed (Ref. 5.4-18).

5.4.1.1.2 Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, i.e., an electric furnace with vacuum degassing. Each plate is fabricated from SA-533, Grade B, Class 1 steel. Flywheel blanks are cut from SA-533, Grade B, Class I plates with at least I/2 in. of stock remaining on the outer surface and bore surface for machining to final dimensions. All welding, including tack welding and repair welding should be prohibited for the flywheels.

Finished machined bore, keyways, and drilled holes are subjected to magnetic particle or liquid penetrant examination in order to meet the requirements of Article NB-2545 or NB-2546 of Section III of the ASME Code. Finished flywheels, as well as the flywheel material, are subjected to 100% volumetric ultrasonic inspection in accordance with procedures and acceptance standards specified in Article NB-2530 of Section III of the ASME Code (Ref. 5.4-14).

The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel ~~should~~ will be inspected for critical dimensions after the over speed test so that any dimensional changes can be detected. ~~Qualified~~ With respect this test procedure, ~~and the acceptance criteria~~ it should be decided ~~with respect to this~~ qualified test procedure and the acceptance criteria.

Flywheels are inspected by a program based on the recommendations of RG 1.14, which references Section XI of the ASME Code (Ref. 5.4-9, 15). The inspection program is discussed in Technical Specification 5.5.7, Reactor Coolant Pump Flywheel Inspection Program and Technical Report “Justification for 20 Years Inspection Interval for Reactor Coolant Pump Flywheel” (Ref. 5.4-23).

5.4.1.1.3 Material Acceptance Criteria

RCP motor flywheels conform to the following material acceptance criteria:

- Nil ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests which exhibit no-break performance at 20°F in accordance with ASTM E-208. The tests prove that the NDTT of the flywheel material does not exceed 10°F.

- A minimum of three charpy v-notch (CVN) impact specimens from each plate are tested at ambient (70°F) temperature in accordance with ASME SA-370 specifications. The CVN energy in both the parallel and normal orientation with respect to the final rolling direction of the flywheel plate material is at least 50 ft-lb and 35-mil lateral expansion at 70°F, and therefore, the flywheel material has a reference nil ductility temperature (RT_{NDT}) of 10°F. An evaluation of the flywheel overspeed proves that integrity of the flywheel is maintained.

5.4.1.2 Reactor Coolant Pump Design Bases

The RCP is in the reactor containment and ensures adequate reactor cooling flow rate to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit that is evaluated in the safety analysis.

The RCP is designed, fabricated, and tested according to the requirements of 10CFR50, 50.55a, GDC 1 and ASME code, Section III (Ref. 5.4-7, 14). The pump is designed with the margin in integrity and exhibits safe operation under all postulated events.

In the event of loss of offsite power (LOOP), the pump is able to provide adequate flow rate during coastdown conditions because of the pump assembly rotational inertia which is provided by the flywheel (top of the motor), the motor rotor, and other rotating parts. This forced flow and the subsequent natural circulation effect in the reactor coolant system (RCS) adequately cools the core.

5.4.5 Reserved by NRC as per RG 1.206

This section is reserved by NRC as per RG 1.206.

5.4.6 Reactor Core Isolation Cooling System

This section is for BWRs only and not applicable to US-APWR.

5.4.7 Residual Heat Removal System

The residual heat removal (RHR) system transfers heat from the RCS to the essential service cooling water system through the component cooling water system (CCWS) to reduce the temperature of the reactor coolant during normal plant shutdown and cool down conditions.

The RHRS is also used to transfer refueling water between the refueling cavity and the refueling water storage pit (RWSP) at the beginning and end of the refueling operations.

5.4.7.1 Design Bases

The RHR system is designed to perform the following functions:

- The RHRS pressure boundary and pressure boundary components are designed to meet GDC 2, GDC 4, GDC 5, GDC 19 Control Room, GDC 34, and BTP 5-4.

- The RHRS is designed to cool the reactor by removing fission products, decay heat, and other residual heat from the reactor core and the RCS after the initial phase of the normal plant shutdown and cool down. Heat is transferred from the RCS through the SGs during the initial cooldown phase.

- The RHRS is designed to ensure that the reactor core decay heat and other residual heat are safely removed from the reactor using four independent subsystems. Any two of the four subsystems have a 100% capability for safe shutdown.

- Each containment spray (CS)/RHR pump and motor-operated valves receive electrical power from safety buses so that the RHRS safety functions are maintained during a loss of offsite power (LOOP).

- Each CS/RHR pump and isolation valve of one train is connected to a separate electrical train so that the RHRS safety functions are maintained during single failure of ~~the~~ an electrical train. This prevents the loss of two or more trains during an electrical failure.

- Assuming the four units of the CS/RHR heat exchangers and four units of the CS/RHR pumps are in service and the supplied essential service water temperature to the component cooling water heat exchanger is at 95°F, the RHRS is designed to provide the capability of reducing reactor coolant temperature during a normal shutdown as follows:

- Reduce reactor coolant temperature to 140°F within 24 hours after reactor shutdown
- Reduce reactor coolant temperature to 130°F within 45 hours after reactor shutdown
- Reduce the reactor coolant temperature to 120°F within 90 hours after reactor shutdown.

Assuming that two units of the CS/RHR heat exchangers and two units of the CS/RHR pumps are in service (assuming a single failure in one train and a second train being out-of-service for preventive maintenance, testing or postulated accident) and the component cooling water (CCW) heat exchanger are supplied with the essential service water temperature at 95°F, the RHRS is capable of reducing the reactor coolant temperature from 350°F to 200°F within 36 hours of the reactor shutdown with two subsystems. A failure modes and effect analysis is provided in Table 5.4.7-1.

- The RHRS is designed to be isolated from the RCS during normal operation. The RHRS is provided with isolation valves in each suction line with interlock capabilities to prevent them from being opened to the RCS above the pressure setpoint.

- The RHRS is designed to remove the fission products decay heat and other residual heat at a rate that assures that the acceptable fuel design limit and the design condition of RCPB are not exceeded.

- The RHRS is designed to provide a portion of the RCS flow to the CVCS during normal plant startup and cooldown operations to control RCS pressure.

- The RHRS is designed to transfer borated water from the RWSP to the refueling cavity at the beginning of a refueling operation. The refueling operation is initiated at a temperature not greater than 140°F. After refueling, the refueling cavity is drained by pumping the water back or by gravity draining to the RWSP.

- The RHRS is designed to provide cooling for the in-containment RWSP during normal plant operations when required. The system is manually initiated by the operator. The RHRS limits the RWSP water temperature to not greater than 120° F during normal operation.

- The RHRS is designed and equipped with pressure relief valves to prevent RHRS over-pressurization and low temperature over-pressurization for RCS components caused by transients, loss of equipment and possible operator error during plant startup, shutdown, and cold shutdown decay heat removal (Refer to Subsection 5.2.2).

- The RHRS is designed for a single nuclear power unit and is not shared between units.

- The RHRS trains are supplied by separate electrical trains thereby being operationally independent of each other.

- The RHRS is designed to be fully operable by the control room operator during single failure for normal operation except for restoring power to the suction isolation valves.

- The RHRS is designed to be operated during mid-loop or drain down operation to allow maintenance or inspection of the reactor head, SG and reactor coolant pump seals.

- The RHR system is designed to prevent an interfacing system LOCA by two motor operated valves in series on the suction line with power lockout capability and design pressure with sufficient wall thickness to withstand the RCS pressure without rupture. In the event that both these valves are opened, the RHR system is designed to withstand high pressure and discharge the RCS inventory to the in-containment RWSP.

- The CS/RHR pump damage from overheating and loss of flow is prevented by minimum flow lines.

- The RHRS is provided with a leakage detection system to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.

- The RHRS is designed for protection against missiles (Refer to Section 3.5), protection against dynamic effects associated with the postulated rupture of piping and pipe whipping (Refer to Section 3.6), discharging fluids inside and outside the containment (Refer to Section 3.4), fires, loss-of-coolant accidents loads, seismic effects (Refer to Section 3.7), and to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (Refer to Section 3.11).

5.4.7.2 System Design

5.4.7.2.1 System Description

The RHRS as shown in Figure 5.4.7-1 consists of four independent subsystems, each of which receives electrical power from one of four safety buses. The P&ID for the four loop RHR system is as shown in Figure 5.4.7-2. Each subsystem includes one CS/RHR pump, one CS/RHR heat exchanger, associated piping, valves, and instrumentation necessary for operational control. Table 5.4.7-2 provides the important system design parameters and CS/RHR pump characteristic curve is as shown in Figure 5.4.7-3. The CS/RHR heat exchangers and the CS/RHR pumps have functions in both the CS system and the RHRS. The mode diagram is as shown in Figure 5.4.7-4.

The RHRS is placed in operation when the pressure and temperature of the RCS are approximately 400 psig and 350° F, respectively.

The inlet lines to the RHRS are connected to the hot legs of four reactor coolant loops (RCLs), while the return lines are connected to the cold legs of each of the RCLs. The RHRS suction lines are isolated from the RCS by two normally-closed motor-operated valves with power lockout capability that are connected in series and located inside the containment. Each discharge line is isolated from the RCS by two check valves located inside the containment and by a normally closed motor-operated valve.

During RHRS operation, each CS/RHR pump takes suction from one of the RCS hot legs by a separate suction line. The pumps then discharge the reactor coolant through the CS/RHR heat exchangers, which transfers heat from the hot reactor coolant fluid to the CCWS circulating through the shell side of the CS/RHR heat exchangers. The cooled reactor coolant is then returned to the RCS cold legs.

CS/RHR pumps transfer borated water from the RWSP to the refueling cavity at the beginning of the refueling operation, and after refueling drain the water back to the RWSP until the water level is brought back to the flange of the reactor vessel.

Coincident with operation of the RHRS, a portion of the reactor coolant flow from the two trains may be diverted downstream of the residual heat exchangers to the CVCS low-pressure isolation letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the CS/RHR heat exchangers. The flow control valve in the bypass line around two of the four CS/RHR heat exchangers automatically maintains a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature, and total flow.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the process sampling system to extract samples from the flow of reactor coolant downstream of the residual heat removal heat exchanger. A local sampling point is also provided on each RHR train between the pump and heat exchanger.

Pump protection is provided using a minimum flow line with an open valve and an orifice at the downstream side of the CS/RHR heat exchanger to the upstream side of the CS/RHR pump suction. This line is sized to provide sufficient pump flow when the CS/RHR pump is shutoff. Also following a LOCA the RHR system is isolated from the RCS and the CSS is placed in operation to provide the make-up source from the RWSP. This is further discussed in Subsection 6.2.2.

The RHRS suction isolation valves in each inlet line from the RCS are interlocked to prevent them from being opened when the RCS pressure is greater than approximately 400 psig. The valves have a control room alarm which alerts the operators if one or both of the valves is not fully closed and the RCS pressure exceeds 400 psig.

5.4.7.2.2 Equipment and Component Description

All components in contact with borated water such as piping, pumps, valves, and equipment for the RHRS are made of austenite stainless steel. However, the shells of the CS/RHR heat exchangers are made of carbon steel. The materials are used to fabricate RHRS components and are in compliance with ASME Section III Materials requirements.

5.4.7.2.3 System Operation

5.4.7.2.3.1 Plant Startup

During the initial stage of the plant startup, the RCS is completely filled with water. Plant startup includes bringing the reactor from the cold shutdown condition to no-load operating temperature and pressure and subsequently to power operation. Generally, while in the cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load present at the time.

At the beginning of plant startup, at least one CS/RHR pump is operating, and the RHRS is aligned to the RCS to divert a portion of the RHR flow through a low pressure letdown path to the CVCS to control the RCS pressure. After the reactor coolant pumps are started, the RHRS is operated as necessary for heat removal. Once the pressurizer steam bubble formation is complete, the RHRS ~~would be~~is isolated from the RCS.

5.4.7.2.3.2 Normal Operation

CS/RHR pumps are not in-service during the normal operation. Normal operation includes the power generation and hot standby operation phases. During normal operation the RHRS is not used and the CS system is on standby. The CS/RHR pumps are normally aligned to take the suction from the RWSP. The tubes of the CS/RHR heat exchangers are filled with borated water and the shells of the heat exchangers are filled with CCW.

5.4.7.2.3.3 Plant Shutdown

Plant shutdown is the operation that brings the reactor plant from normal operating temperature and pressure to refueling condition. The initial phase of plant shutdown is accomplished by transferring heat from the RCS to the steam and power conversion system through SGs. Depressurization is accomplished by spraying reactor coolant into the pressurizer which cools and condenses the pressurizer steam bubble.

The second phase of cooldown starts with the RHRS being placed in operation when the reactor coolant temperature and pressure are reduced to approximately 350°F and 400 psig, respectively, approximately four hours after reactor shutdown. Startup of the RHRS includes a warm up period, during which time reactor coolant flow rate is slowly increased through the heat exchangers to protect the piping/components in the RHR system from thermal shock.

The rate of heat removal from the reactor coolant is manually controlled by the operator by regulating the coolant flow through the CS/RHR heat exchanger. This is accomplished by re-opening the CS/RHR heat exchanger outlet flow control valves in two subsystems. The CS/RHR heat exchanger outlet flow control valves are positioned by the operator who maintains the total flow rate constantly through the CS/RHR heat exchanger bypass-flow control valves.

oxidation operation and installation/removal of steam generator (SG) nozzle cover. When the RCS water level decreases abnormally, air inadvertently gets into the residual heat removal system with the possibility of affecting the RHRS.

The features of mid-loop operation in US-APWR are shown as follows;

A. Chemical addition (hydrogen peroxide)

Hydrogen peroxide addition is adopted instead of aeration because it decreases the duration of the mid-loop operation. As a result, the mid-loop operation is needed only to drain the SG primary side water while being able to maintain a high RCS water level for most of the oxidation operation.

B. High RCS water Level

Keeping a high RCS water level for the duration of SG primary side water drainage and vacuum venting operation decreases the possibility of air intake to the RHRS. Since the SG installation level for the US-APWR is higher than in most plants, a high RCS water level during the oxidation operation does not affect the SG nozzle cover, nor interfere with the SG maintenance.

C. ~~Redundant w~~Water level instrument

Redundant narrow range water level instrument and a mid-range water level instrument, which are shown in Figure 5.1-2 (Sheet 3 of 3), are provided to measure mid-loop water level. Installation of a redundant water narrow level instrument enhances reliability of the mid-loop operation.

A temporary mid-loop water level sensor that measures the RCS water level with reference to pressure at the reactor vessel head vent line and cross over leg is installed in addition to these permanent water level sensors to cope with surge line flooding events.

D. Interlock for abnormal water level decrease

When the water level of RCS drops below the mid-loop level, low pressure letdown lines are isolated automatically. This interlock is useful to prevent loss of reactor coolant inventory

E. Water supply from spent fuel pit

When the water level of RCS abnormally drops and all RHR pumps cannot be operated because of air intake, operator can supply water from the spent fuel pit (SFP) to the reactor vessel. Since the RHRS is connected to the SFP, SFP water can be injected by gravity.

The level in the primary system is lowered below the upper end of the hot and cold legs. The RCS water level should be maintained higher than 0.33 feet above the loop center

The design stress limits, loads, and combined loading conditions are discussed in Subsection 3.9.3.

Design transients for the components of the pressurizer are discussed in Subsection 3.9.1. The pressurizer surge nozzle and surge line are designed to withstand the thermal stresses resulting from volume surges occurring during operation.

The pressurizer is designed in accordance with the requirements of GDC 2 and GDC 4 (Ref. 5.4-8).

5.4.10.2 System Description

5.4.10.2.1 Pressurizer

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads. It is constructed of low-alloy steel and clad with austenitic stainless steel on the internal surfaces in contact with the reactor coolant. The pressurizer pressure boundary materials are described in Subsection 5.2.3. The pressurizer material specifications are shown in Table 5.2.3-1.

The pressurizer configuration is shown in Figure 5.4.10-1. The pressurizer design data are provided in Table 5.4.10-1. Codes and material requirements are provided in Section 5.2.

The safety valve nozzles, spray nozzle and safety depressurization valve nozzle are located at the top head. Spray flow is modulated by automatically controlled air-operated valves. The spray valves can also be operated manually from the main control room. The surge nozzle is located at the bottom head. Both the spray and surge nozzles are provided with thermal sleeves for protection against thermal transients.

The surge nozzle located at the bottom head is provided with a surge screen mounted above the nozzle to prevent ingress of foreign objects from the pressurizer to the RCS. To restrict direct in-surge flow of cold water to the steam/water interface and for mixing purposes, guide plate(s) are provided above the surge nozzle. The guide plate(s) also serves as heater supports.

Electric immersion heaters are vertically installed through the bottom head. The heater sheath is welded on a heater sleeve end protruding externally from the bottom head. The heater sleeve is welded at the inner surface of the bottom head as a pressure retaining part.

The heaters are grouped into a control group and backup groups. The heaters in the control group are proportional heaters which are supplied with continuously variable power to match heating needs. The heaters in the backup group are either off or at full power. The power supply to the heaters is a 480-volt 60 Hz three-phase circuit. Each heater is connected to one leg of a delta-connected circuit and is rated at 480 volts with one-phase current. The capacity of the control and backup groups is defined in Table 5.4.10-2. At least 120 kW capacity is required for the heaters in the backup groups A, B, C and D each to maintain the RCS pressure near normal operating pressure.

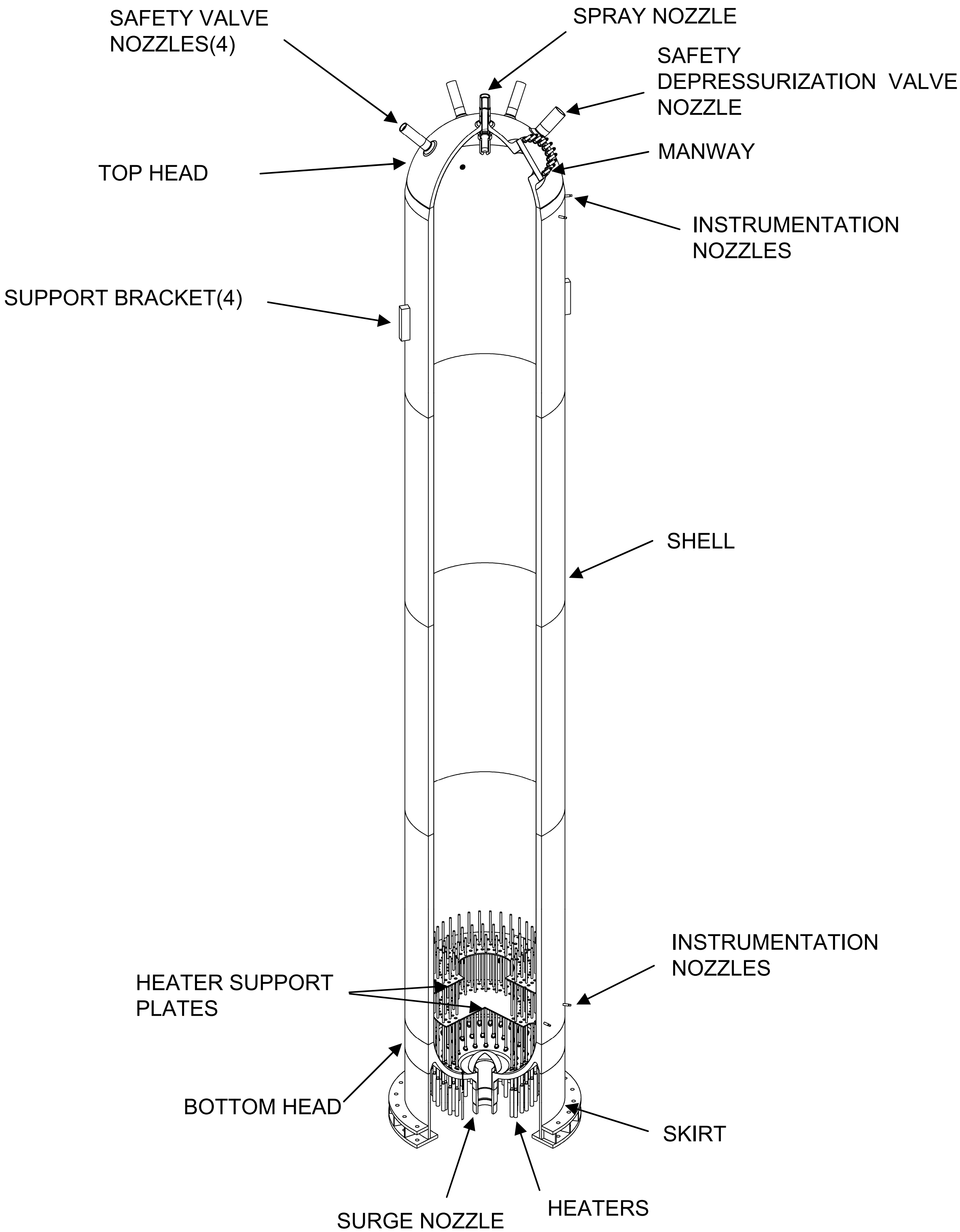

Figure 5.4.10-1 Pressurizer

Table 5.4.12-1 Reactor Vessel Head Vent Design Parameters

System design pressure (psig)	2,485
System design temperature (°F)	650
Piping diameter (in)	1 (schedule 160)
Hydrogen gas discharging capacity at 1,200 psia (lbm/~~h~~sec)	0.43

Table 5.4.12-2 Safety Depressurization Valve Design Parameters

Type	Motor operated valve
System design pressure (psig)	2,485
System design temperature (°F)	680
Number	2
Saturated steam discharging capacity at 2,335 psig (lb/h)	530,000

Table 5.4.12-3 Depressurization Valve Design Parameters

Type	Motor operated valve
System design pressure (psig)	2,485
System design temperature (°F)	680
Number	2
Saturated steam discharging capacity at 2335 psig (lb/h)	795,000

Chapter 6

US-APWR DCD Chapter 6 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
6-xix	Acronyms and Abbreviations	Deleted “CIS” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.0-2	6.0.2	Deleted “CIS” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.0-3	6.0.3 third paragraph	Correction Replaced “two reactor vessel hot legs (B and D)” to “two reactor vessel hot legs (A and D)”.
6.2-43	6.2.2.2	Sentencees “The CSS containment isolation valves are interlocked and are allowed to open only if two in-series RHR hot leg suction isolation valves are closed” was replaced with “The CSS containment isolation valves are interlocked and are allowed to open only if either of the corresponding two in-series RHR hot leg suction isolation valve is closed.” Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.2-55 to 56	6.2.4.3	“CIS” was replaced with “containment isolation system” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.2-90 to 92	Table 6.2.1-17 Sheet 1 and 2 of 2	The title “ LBB / Other considerations” was replaced with “Results of LBB Evaluation” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.2-194	Table 6.2.4-3	Editorial The row of RCS-VLV-171 is deleted.
6.2-200	Table 6.2.4-3	Editorial IAS-MOV-002, Actuation signal is changed to “T” from “P”.
6.2-201	Table 6.2.4-3	Editorial Pen NO. P409, added the valve, VWS-VLV-426, and related information.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
6.2-210	Table 6.2.4-3	Editorial “S signal” was changed to “Containment Isolation Phase A” . “CV spray signal” was changed to “Containment Isolation Phase B”. “Containment Ventilation Isolation Signal” was changed to “Containment Purge Isolation Signal”. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.2-299	Figure 6.2.2-5	Other Deleted platform elevation.
6.2-306	Figure 6.2.4-1 (Sheet 3 of 52)	Editorial Deleted the valve (RCS-VLV-171), and added penetration number(PEN#260) .
6.2-337	Figure 6.2.4-1 (Sheet 34 of 52)	Editorial Added the valve number (DWS-VLV-006).
6.2-346	Figure 6.2.4-1 (Sheet 43 of 52)	Editorial Added “LC” on VWS-VLV-426.
6.2-355	Figure 6.2.4-1 (Sheet 52 of 52)	Editorial Added penetration number “PEN#419”.
6.3-7	6.3.2.2.3 first paragraph	Editorial Replaced “47,680” with “29,410”.
6.3-16	6.3.2.8 fifth paragraph	Correction Replaced “the reactor hot leg B or D” with “the reactor hot leg A or D”.
6.4-6	6.4.2.2.3 6.4.2.2.4	Editorial “The above mentioned isolation dampers are Equipment Class 3, seismic category I components.” was added in 6.4.2.2.3. Section “6.4.2.2.4 Shutoff Dampers” was added. “isolation” was replaced with “shutoff” in section 6.4.2.2.4. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
6.4-11	Table 6.4-1	“(Elemental and Organic)” was added to the description. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
6.5-1	6.5.1 second paragraph	RAI No.670, Question No. 09.04.05-16 Added "The penetration areas and the safeguard component areas are shown in Figure 6.5-2 through 6.5-9." in the second paragraph.
6.5-4	6.5.1.3	"motor operated" was replaced with "electro hydraulic operated". Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
New page (6.5-23 through 6.5-30)	Figure 6.5-2 through 6.5-9	RAI No.670, Question No. 09.04.05-16 Added new figures, Figure 6.5-2 through 6.5-9.
6.6-1	6.6.1 first paragraph	Other Following sentence is added, "The preservice inspection and ISI of threaded fasteners, in accordance with the requirements and the criteria of ASME Code, Section XI for bolting and mechanical joints used in ASME Code Class 2 systems, is described in Subsection 3.13.2."
6.6-3	6.6.3 second paragraph	Other Deleted "ultrasonic, " in the last sentence of the second paragraph.
6.6-4	6.6.5 first paragraph	Other Added "as modified by 10CFR50" on the last sentence of 1st paragraph.
6.6-5	6.6.8 first paragraph	Other Replaced "Category C-f" with "Categories C-F-1 and C-F-2"

ACRONYMS AND ABBREVIATIONS

ac	alternating current
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BAT	boric acid tank
BBR	BBR VT International Ltd
BWR	boiling water reactor
BWROG	boiling water reactor owners' group
CCWS	component cooling water system
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CHS	containment hydrogen monitoring and control system
~~CIS~~	~~containment isolation system~~
COL	Combined License
CRE	control room envelope
CSS	containment spray system
C/V	containment vessel
CVCS	chemical and volume control system
CVDT	containment vessel reactor coolant drain tank
CVTR	Carolinas-Virginia Tube Reactor
DBA	design-basis accident
dc	direct current
DCD	Design Control Document
DECLG	double-ended cold leg guillotine pump discharge
DEGB	double-ended guillotine break
DEHLG	double-ended hot leg guillotine
DEPSG	double-ended pump suction guillotine
DF	decontamination factor
DNBR	departure from nucleate boiling ratio
DOP	dioctyl phthalate
DPS	containment depressurization system
DVI	direct vessel injection
EAB	exclusion area boundary
ECCS	emergency core cooling system
EFW	emergency feedwater
ERDA	Energy Research and Development Administration (now U.S. DOE)
ESF	engineered safety features
ESFAS	engineered safety feature actuation system

requirements, and demonstrates how the ESF design meets or exceeds the functional requirements. This section of the Design Control Document (DCD) lists and discusses each system that is considered to be part of the ESF systems.

6.0.1 Engineered Safety Feature Material

The materials used in constructing and fabricating ESF components and systems, as well as their interaction with ECCS fluids and post-accident conditions, are considered in Section 6.1. The material specifications, selection, treatment, and coatings are described. Materials are selected and treated to improve hardness, strength, corrosion resistance, and ductibility; and to reduce the probability of a rapidly propagating fracture.

6.0.2 Containment Systems

The US-APWR containment, as discussed in Subsection 6.2.1, completely encloses the reactor and RCS. The containment is essentially leak tight to ensure that no significant amount of radioactive material can reach the environment, even in the unlikely event of a RCS failure.

The containment is a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat, reinforced concrete foundation slab. To ensure leak tightness during normal operation and under postulated accident conditions, the US-APWR containment is designed and built to safely accommodate an internal pressure of 68 psig.

The following are US-APWR containment systems:

- containment heat removal system
- containment isolation system~~ (CIS)~~
- containment hydrogen monitoring and control system (CHS)

The containment spray system (CSS) limits the peak containment pressure to less than the design pressure and is capable of reducing the containment pressure to approximately atmospheric in the unlikely event of an accident. The CSS shares the residual heat removal system (RHRS) pumps and heat exchangers. The containment spray piping, spray rings, and nozzles are unique to the CSS.

All lines that penetrate the containment are provided with isolation features. The containment isolation system valves that automatically close when required do not automatically re-open when the isolation condition “clears.” If a loss of actuating power occurs, the valves remain closed. Re-opening such automatic containment isolation valves requires deliberate, manual action by a plant operator.

The CHS monitors and limits the concentration of hydrogen in containment. In the unlikely event that excessive hydrogen is detected in containment, hydrogen igniters burn excess hydrogen in a controlled manner, thus, avoiding potential, localized containment damage.

The US-APWR containment is designed to permit periodic leakage rate testing. The periodic leakage rate testing program is the responsibility of any utility that references the US-APWR design for construction and licensed operation.

6.0.3 Emergency Core Cooling Systems

The ECCS removes heat from the reactor core following postulated design basis events. The US-APWR ECCS consists of the following:

- accumulator system
- high head injection system
- emergency letdown system

The accumulators are passive devices that inject borated water directly into each of four reactor cold legs. The accumulators have a dual flow rate design; a large initial flow rate for the immediate vessel refill, and a small flow rate of longer duration for a continued core re-flood.

The high head injection system combines its flow performance with the flow rate of the accumulators to ensure a timely flow response and a long-term injection for core cooling. The safety injection pumps automatically start and deliver borated water from the refueling water storage pit for the duration of the event. Four, 50% capacity, safety injection pumps are provided.

The emergency letdown system performs a “feed and bleed” (FAB) letdown boration to establish cold shutdown conditions if the normal chemical and volume control system (CVCS) is unavailable. The emergency letdown system directs the reactor coolant from two reactor vessel hot legs (~~B~~A and D) to the refueling water storage pit, from which highly borated water can be returned to the reactor vessel using the safety injection pumps.

6.0.4 Habitability Systems

The control room habitability system is the ESF that allows operators to remain safely inside the control room envelope while taking the necessary actions to manage and control unusual, unsafe, or abnormal plant conditions, including a loss-of-coolant accident (LOCA). The control room habitability system protects the operators against postulated releases of radioactive material, toxic gases, and smoke, and enables the operators to occupy the control room envelope safely and for an extended time.

6.0.5 Fission Product Removal and Control Systems

Fission product removal systems are ESFs that confine fission products that are released from the reactor core as a result of the design basis LOCA and become airborne. Sometimes referred to as “atmosphere cleanup,” fission products are confined in the sense that their free mobility and circulation would otherwise raise the potential of an unintended release to the environment. The containment controls reduce leakage of fission products from the containment to ensure that the leakage fraction that may reach environmental conditions. All valves required to be actuated during CSS operation are located to prevent vulnerability to flooding.

Protection of the CSS from missiles is discussed in Section 3.5. Protection of the CSS against dynamic effects associated with rupture of piping is described in Section 3.6. Protection from flooding is discussed in Section 3.4.

The CSS is designed for periodic inservice testing and inspection of components in accordance with ASME Code Section XI.

6.2.2.2 System Design

Figure 6.2.2-1 is the flow diagram of the CSS, showing the major components, instruments, and the appropriate system interconnections. Table 6.2.2-1 presents design and performance data for CSS components. The performance data for CS/RHR pump and CS/RHR heat exchanger is shown in Chapter 5, Subsection 5.4.7.

The CSS receives electrical power for its operation and control from onsite emergency power sources and offsite sources, as shown in Chapter 8. In the unlikely event of a LOCA or secondary system line break that significantly increases the containment pressure, the containment spray automatically initiates to limit peak containment pressure to well below the containment design pressure. In addition to preserving containment structural integrity, containment spray limits the potential post-accident radioactive leakage by reducing the pressure differential between the containment atmosphere and the environment.

The CS/RHR system can be manually initiated and operated from the MCR and the remote shutdown console (RSC). In addition to the typical system status and operating information (e.g., valve position indication, pump run status), the containment temperature and pressure are indicated and recorded in the MCR and RSC.

Dual-use components are the CS/RHR heat exchangers and CS/RHR pumps. Motor-operated valves permit CSS or RHRS recirculation of the reactor core. The four CSS containment isolation valves are normally closed, but open automatically on a P signal. ~~The CSS containment isolation valves are interlocked and are allowed to open only if two in-series RHR hot leg suction isolation valves are closed~~The CSS containment isolation valves are interlocked and are allowed to open only if either of the corresponding two in-series RHR hot leg suction isolation valve is closed. Further, the RHR hot leg suction valves are interlocked so that they cannot be opened unless the corresponding CSS containment isolation valves are closed. This arrangement prevents the reactor vessel water inventory from being sprayed into the containment.

Following a DBA, the containment pressure approaches atmospheric pressure. When the containment pressure is reduced sufficiently and the operator determines that containment spray is no longer required, the operator terminates containment spray. The operator closes the containment spray header isolation valves and aligns system flow through the CS/RHR heat exchanger back to the RWSP through the full flow test line. The pit water is then recirculated and cooled.

- Seal water injection
- Post-accident sampling return line
- Fire protection water supply system

The condition in which containment isolation is needed in safety injection system, containment spray system and residual heat removal system is when leak occurs in these systems. These systems are located in safeguard component area. Leak detection system is installed in each system. Level instruments are installed in each pump compartment sump. In addition, if leak is occurred, operators can notice by pump suction/discharge pressure and pump flow rate. As for main steam system, NMS-MOV-507A, B, C, D, NMS-MOV-701A, B, C, D and EFS-MOV-101A, B, C, D are remote manual isolation valves. The condition in which containment isolation is needed is to prevent fission product from releasing such as in SGTR. In each main steam line, radiation monitors is installed. So operators can notice that these valves should be closed. As for seal water injection line, CVS-MOV-178 A, B, C, D are remote manual isolation valves. The condition in which containment isolation is needed is the case that seal injection flow is lost. In each injection line, flow rate instrument is installed. So operators can notice that these valves should be closed. As for post-accident sampling return line and fire protection water supply system, PSS-MOV-071 and FSS-MOV-004 are remote manual isolation valves. The reason why these valves does not receive containment isolation signal is that these are closed under administrative control, such as locked closed. Therefore, these valves are not needed to be closed if leak occur.

Containment purge isolation valves (Containment Purge System) may be supplied with resilient seals and the subject containment penetrations and containment isolation valves will receive preoperational and periodic Type C leak rate testing in accordance with 10 CFR 50, Appendix J. The soft seated containment isolation butterfly valves in the containment purge system which may require resilient seal replacement following the leakage rate testing will be subject to seals replacement based on a valve manufacturer recommendation.

Table 6.2.4-1 presents the design information regarding provisions for isolating the containment penetrations, while Table 6.2.4-2 and Figure 6.2.4-1 presents associated containment isolation configurations. Table 6.2.4-3 presents the list of containment penetrations and system isolation positions, which includes the information related to the pipe length from containment to outermost isolation valve.

6.2.4.3 Design Evaluation

The piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. These piping systems are designed with the capability to test, periodically, the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.

The ~~CIS~~containment isolation system is able to perform its safety function in the event of any single active failure. The ~~CIS~~containment isolation system includes double isolation barriers at the containment penetrations. Redundant isolation valves are powered from separate electrical trains to provide containment isolation in the event of a single active failure in the electrical system. Therefore, ~~CIS~~containment isolation system meets the single failure criterion.

6.2.4.3.1 Evaluation of Conformance to General Design Criterion 55 of 10CFR50, Appendix A

Each line that is part of the RCPB and penetrates containment is provided with containment isolation valves, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. Isolation valves outside containment are located as close to containment as practical for those systems designed in conformance with GDC 55 or some other defined basis set forth in RG 1.141. The following systems penetrating containment meet GDC 55 criteria:

- SIS N_2 supply line to the accumulators, the RHRS return line, and the primary makeup water system (PMWS) demineralized water supply line, using one automatic isolation valve inside and one locked closed isolation valve outside the containment.
- RCS PMWS line to the PRT using three valves, one automatic isolation valve and one locked closed manual isolation valve inside and one automatic isolation valve outside containment.
- CVCS letdown line/charging line/seal injection line for RCPs/seal water return line, SIS SI line, process and post accident sampling system (PSS) pressurizer gas and liquid phase sampling line, in core instrument gas purge system (ICIGS) CO_2 line, waste management system (WMS) reactor coolant drain tank gas analysis line/N_2 supply and vent line/pump discharge line, accumulator sample line, and the RCS N_2 supply line to the pressurizer relief tank (PRT) using one automatic isolation valve inside and one automatic isolation valve outside the containment.

Containment isolation provisions for lines in ESF or ESF-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if the system reliability can be shown to be greater, the system is closed outside the containment, and a single active failure can be accommodated with only one isolation valve in the line. Table 6.2.4-2 lists GDC 55 systems with single valve isolation and justification, in accordance with the guidance in NUREG-0800, SRP 6.2.4 (Ref. 6.2-27).

6.2.4.3.2 Evaluation of Conformance to General Design Criterion 56 of 10CFR50, Appendix A

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment is provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. Isolation valves outside containment are located as close to containment as practical for those systems designed in conformance with GDC 56 or some other defined basis set forth in RG 1.141. The following systems penetrating the containment meet GDC 56 criteria:

Table 6.2.1-17 Subcompartment and Postulated Break Line condition (Sheet 1 of 2)

Subcompartment	Break Line	Line Spec	Press. psi	Temp. °F	fluid	~~LBB / Other considerations~~Results of LBB Evaluation
Steam Generator Subcompartment	Main Coolant Pipe-Hot Leg	31"ID-RCS-2501	2235	617.0	Subcooled Water	Leak
	Main Coolant Pipe-Cold Leg	31"ID-RCS-2501	2235	550.6	Subcooled Water	Leak
	Main Coolant Pipe-Cross-over Leg	31"ID-RCS-2501	2235	550.6	Subcooled Water	Leak
	Pressurizer Surge Line	16"-RCS-2501	2235	653.0	Saturated Water	Leak
	Accumulator Injection Line	14"-RCS-2501	2235	550.6	Subcooled Water	Leak
		14"-SIS-2501	2235	550.6	Subcooled Water	Leak
		14"-SIS-2511	2235	120.0	Subcooled Water	Leak
	RHR Pump Inlet Line	10"-RCS-2501	2235	617.0	Subcooled Water	Break
	RHR Pump Outlet Line	8"-RCS-2501	2235	550.6	Subcooled Water	Break
	Direct Vessel Injection Line	4"-RCS-2501	2235	550.6	Subcooled Water	Break
	SI High Head Injection Line	4"-RCS-2501	2235	617.0	Subcooled Water	Break
	SI Emergency Letdown Line	2"-RCS-2501	2235	617.0	Subcooled Water	Break
	Pressurizer Spray Line	6"-RCS-2501	2235	550.6	Subcooled Water	Break
	Loop Drain Line	2"-RCS-2501	2235	550.6	Subcooled Water	Break
	Charging Line	4"-RC-2501	2235	550.6	Subcooled Water	Break
		4"-CVS-2501	2235	550.6	Subcooled Water	Break
		4"-CVS-2561	2235	464.0	Subcooled Water	Break
	Letdown Line	3"-RCS-2501	2235	550.6	Subcooled Water	Break
		3"-CVS-2501	2235	550.6	Subcooled Water	Break
		3"-CVS-2561	2235	550.6	Subcooled Water	Break
		3"-CVS-0601	350	269.1	Subcooled Water	Break
		4"-CVS-0601	350	115.0	Subcooled Water	Break
	RCP Seal Water Injection Line	1-1/2"-CVS-2501	2600	130.0	Subcooled Water	Break
		1-1/2"-CVS-2511	2600	130.0	Subcooled Water	Break
	Feedwater Line	16"-FWS-1525	1185	568.0	Saturated Water	Break
	Main Steam Line	32"-MSS-1532	907	535.0	Steam	Leak
	SG Blowdown Line	3"-SGS-1532	907	535.0	Steam	Break

Subcompartment	Break Line	Line Spec	Press. psi	Temp. °F	fluid	~~LBB / Other considerations~~Results of LBB Evaluation
		4”-SGS-1532	907	535.0	Steam	Break

Table 6.2.1-17 Subcompartment and Postulated Break Line condition (Sheet 2 of 2)

Subcompartment	Break Line	Line Spec	Press. psi	Temp. °F	fluid	~~LBB / Other considerations~~Results of LBB Evaluation
Subcompartment under Pressurizer Subcompartment	Pressurizer Surge Line	16”-RCS-2501	2235	653.0	Saturated Water	Leak
Pressurizer Subcompartment	Pressurizer Spray Line	6”-RCS-2501	2235	550.6	Subcooled Water	Break
	Pressurizer Auxiliary Spray Line	3”-RCS-2501	2235	550.6	Subcooled Water	Break
	Pressurizer Safety Valve Inlet Line	6”-RCS-2501	2235	653.0	Steam	Break
	Pressurizer Safety Depressurization Line	8”-RCS-2501	2235	653.0	Steam	Break
		6”-RCS-2501	2235	653.0	Steam	Break
		4”-RCS-2501	2235	653.0	Steam	Break
Pressurizer spray valve room	Pressurizer Spray Line	6”-RCS-2501	2235	550.6	Subcooled Water	No Break
Reactor Cavity	Direct Vessel Injection Line	4”-RCS-2501	2235	554.6	Subcooled Water	Break
Regenerative heat exchanger room	Charging Line	4”-CVS-2511	2600	130.0	Subcooled Water	Break
		4”-CVS-2561	2266	554.6	Subcooled Water	Break
	Letdown Line	3”-CVS-2561	2266	554.6	Subcooled Water	Break
Regenerative heat exchanger valve room	Charging Line	4”-CVS-2561	2366	554.6	Subcooled Water	Break
	Letdown Line	3”-CVS-2561	2266	380	Subcooled Water	Break
		3”-CVS-0601	350	380	Subcooled Water	Break
Letdown heat exchanger room	Letdown Line	4”-CVS-0601	350	380	Subcooled Water	Break

Table 6.2.4-3 List of Containment Penetrations and System Isolation Positions (Sheet 1 of 12)

Pen NO.	GDC	System Name	Fluid	Line Size (in.)	ESF or Support System	Valve Arragmt Figure 6.2.4-1	Valve Number	Location of Valve	Type Tests	Type C Test	Length of Pipe (Note 1)	Valve: Type	Valve: Operator	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time	Power Source (Note 2)	Remark
P247	56	RCS	Nitrogen Gas	1	No	Sht. 2	RCS-VLV-133	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1			RCS-AOV-132	Out			9.0 ft	Dia	Air	Auto	RM	O	C	C	FC	T	15	1E	
				3/4			RCS-VLV-167	In			-	Dia	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P260	56	RCS	Demi. Water	3	No	Sht. 3	RCS-VLV-139	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				3			RCS-VLV-140	In			-	Dia	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
				3			RCS-AOV-138	Out			10.0 ft	Globe	Air	Auto	RM	O	C	C	FC	T	15	1E	
~~-~~	~~-~~	~~-~~	~~-~~	~~3/4~~	~~-~~	~~-~~	~~RCS-VLV-171~~	~~In~~	~~-~~	~~-~~	~~-~~	~~Dia~~	~~Manual~~	~~Manual~~	~~None~~	~~C~~	~~C~~	~~C~~	~~NA~~	~~NA~~	~~NA~~	~~NA~~	
P276L	56	RCS	Nitrogen Gas	3/4	No	Sht. 4	RCS-AOV-147	In	C	Y	-	Globe	Air	Auto	RM	O	C	C	FC	T	15	1E	
				3/4			RCS-AOV-148	Out			10.0 ft	Globe	Air	Auto	RM	C	C	C	FC	T	15	1E	
P277	55	CVCS	Primary Coolant	4	No	Sht. 5	CVS-AOV-005	In	C	Y	-	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	
				4			CVS-AOV-006	Out			14.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	
P278	55	CVCS	Primary Coolant	4	No	Sht. 6	CVS-VLV-153	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				4			CVS-MOV-152	Out			14.0 ft	Gate	Motor	Auto	RM	O	O	C	FAI	S	20	1E	
				3/4			CVS-VLV-653	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P279	56	CVCS	Primary Coolant	1 1/2	No	Sht. 7	CVS-VLV-179B	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1 1/2			CVS-MOV-178B	Out			14.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	15	1E	
				3/4			CVS-VLV-667B	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P280	56	CVCS	Primary Coolant	1 1/2	No	Sht. 7	CVS-VLV-179D	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1 1/2			CVS-MOV-178D	Out			14.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	15	1E	
				3/4			CVS-VLV-667D	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P281	56	CVCS	Primary Coolant	1 1/2	No	Sht. 7	CVS-VLV-179A	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1 1/2			CVS-MOV-178A	Out			14.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	15	1E	
				3/4			CVS-VLV-667A	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P282	56	CVCS	Primary Coolant	1 1/2	No	Sht. 7	CVS-VLV-179C	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1 1/2			CVS-MOV-178C	Out			14.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	15	1E	
				3/4			CVS-VLV-667C	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P283	55	CVCS	Primary Coolant	3	No	Sht. 8	CVS-MOV-203	In	C	Y	-	Globe	Motor	Auto	RM	O	O	C	FAI	P,T+UV	15	1E	
				3			CVS-MOV-204	Out			9.0 ft	Globe	Motor	Auto	RM	O	O	C	FAI	P,T+UV	15	1E	
				3/4			CVS-VLV-202	In			-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
P236	56	SIS	Nitrogen Gas	1	No	Sht. 9	SIS-VLV-115	In	C	Y	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	
				1			SIS-AOV-114	Out			9.0 ft	Globe	Air	Auto	RM	C	C	C	FC	T	15	1E	
				3/4			SIS-VLV-156	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P210	55	SIS	Borated Water	4	Yes	Sht. 10	SIS-VLV-010A	In	~~C~~A	~~Y~~N	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	Note 4
				4			SIS-MOV-009A	Out			9.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	20	1E	
				3/4			SIS-VLV-058A	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P227	55	SIS	Borated Water	4	Yes	Sht. 10	SIS-VLV-010B	In	~~C~~A	~~Y~~N	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	Note 4
				4			SIS-MOV-009B	Out			9.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	20	1E	
				3/4			SIS-VLV-058B	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	
P258	55	SIS	Borated Water	4	Yes	Sht. 10	SIS-VLV-010C	In	~~C~~A	~~Y~~N	-	Check	Self	Auto	None	-	-	-	NA	NA	NA	NA	Note 4
				4			SIS-MOV-009C	Out			9.0 ft	Globe	Motor	RM	Manual	O	O	O	FAI	RM	20	1E	
				3/4			SIS-VLV-058C	In			-	Globe	Manual	Manual	None	C	C	C	NA	NA	NA	NA	

Table 6.2.4-3 List of Containment Penetrations and System Isolation Positions (Sheet 6 of 12)

Pen NO.	GDC	System Name	Fluid	Line Size (in.)	ESF or Support System	Valve Arragmt Figure 6.2.4-1	Valve Number	Location of Valve	Type Tests	Type C Test	Length of Pipe (Note 1)	Valve		Actuation Mode		Valve Position			Power Failure	Actuation Signal	Valve Closure Time	Power Source (Note 2)	Remark
												Type	Operator	Primary	Secondary	Normal	Shutdown	Post-Accident					
P239R	57	SGBDS	Secondary	3/4	No	Sht. 31	SGS-AOV-031C	Out	A	N	11.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	15	1E	Note 5
P239L	57	SGBDS	Coolant	3/4	No	Sht. 31	SGS-AOV-031D	Out	A	N	12.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	15	1E	Note 5
P505	57	SGBDS	Secondary	4	No	Sht. 31	SGS-AOV-001A	Out	A	N	22.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	Note 5
P506	57	SGBDS	Coolant	4	No	Sht. 31	SGS-AOV-001B	Out	A	N	26.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	Note 5
P507	57	SGBDS		4	No	Sht. 31	SGS-AOV-001C	Out	A	N	26.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	Note 5
P508	57	SGBDS		4	No	Sht. 31	SGS-AOV-001D	Out	A	N	22.0 ft	Globe	Air	Auto	RM	O	O	C	FC	T	20	1E	Note 5
P161	56	RWS	Borated Water	6 6 3/4	No	Sht. 32	RWS-MOV-002 RWS-MOV-004 RWS-VLV-003	In Out In	C	Y	- 19.0 ft -	Gate Gate Check	Motor Motor Self	Auto Auto Auto	RM RM None	O O -	O O -	C C -	FAI FAI NA	T T NA	30 30 NA	1E 1E NA	
P162	56	RWS	Borated Water	4 4 3/4	No	Sht. 33	RWS-VLV-023 RWS-AOV-022 RWS-VLV-073	In Out In	C	Y	- 29.0 ft -	Check Dia Globe	Self Air Manual	Auto Auto Manual	None RM None	- O C	- O C	- C C	NA FC NA	NA T NA	NA 20 NA	NA 1E NA	
P253	56	PMWS	Deminrralized Water	2 2 3/4	No	Sht. 34	DWS-VLV-005 DWS-VLV-004 DWS-VLV-006	In Out In	C	Y	- 9.0 ft -	Check Dia Dia	Self Manual Manual	Auto Manual Manual	None None None	- C C	- C C	- C C	NA NA NA	NA NA NA	NA NA NA	NA NA NA	
P245	56	IAS	Compressed Air	2 2 3/4	No	Sht. 35	IAS-VLV-003 IAS-MOV-002 IAS-VLV-004	In Out In	C	Y	- 9.0 ft -	Check Globe Globe	Self Motor Manual	Auto Auto Manual	None RM None	- O C	- O C	- C C	NA FAI NA	NA ~~P~~T NA	NA 15 NA	NA 1E NA	
P248	56	FSS	Fire Water	3 3 3/4	No	Sht. 36	FSS-VLV-003 FSS-AOV-001 FSS-VLV-002	In Out In	C	Y	- 9.0 ft -	Check Globe Globe	Self Air Manual	Auto Auto Manual	None RM None	- C C	- C C	- C C	NA FC NA	NA T NA	NA 15 NA	NA 1E NA	
P238	56	FSS	Fire Water	6 6 3/4	No	Sht. 37	FSS-VLV-006 FSS-MOV-004 FSS-VLV-005	In Out In	C	Y	- 10.0 ft -	Check Gate Globe	Self Motor Manual	Auto Auto Manual	None RM None	- C C	- C C	- C C	NA FAI NA	NA RM NA	NA 30 NA	NA 1E NA	
P230	56	SSAS	Compressed Air	2 2 3/4	No	Sht. 38	SAS-VLV-103 SAS-VLV-101 SAS-VLV-102	In Out In	C	Y	- 9.0 ft -	Check Globe Globe	Self Manual Manual	Auto Manual Manual	None None None	- C C	- C C	- C C	NA NA NA	NA NA NA	NA NA NA	NA NA NA	
P200	-	-	(Fuel Transfer Tube)	22	No	Sht. 39	-	-	B	N	-	Flange	NA	-	-	C	C	C	NA	NA	NA	NA	
P451	56	HVAC	Containment Atmosphere	36 36	No	Sht. 40	VCS-AOV-305 VCS-AOV-304	In Out	C	Y	- 13.0 ft	B-fly B-fly	Air Air	Auto Auto	RM RM	C C	O O	C C	FC FC	V V	5 5	1E 1E	
P452	56	HVAC	Containment Atmosphere	36 36	No	Sht. 40	VCS-AOV-306 VCS-AOV-307	In Out	C	Y	- 9.0 ft	B-fly B-fly	Air Air	Auto Auto	RM RM	C C	O O	C C	FC FC	V V	5 5	1E 1E	
P410	56	HVAC	Containment Atmosphere	8 8	No	Sht. 41	VCS-AOV-356 VCS-AOV-357	In Out	C	Y	- 10.0 ft	B-fly B-fly	Air Air	Auto Auto	RM RM	C C	C C	C C	FC FC	V V	5 5	1E 1E	
P401	56	HVAC	Containment Atmosphere	8 8	No	Sht. 41	VCS-AOV-355 VCS-AOV-354	In Out	C	Y	- 10.0 ft	B-fly B-fly	Air Air	Auto Auto	RM RM	C C	C C	C C	FC FC	V V	5 5	1E 1E	
P262R	56	HVAC	Silicone Oil	3/4	No	Sht. 42	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	Note 8
P262L	56	HVAC	Silicone Oil	3/4	No	Sht. 42	-	-	A	N	-	-	-	-		-	-	-	-	-	-	-	Note 8

Table 6.2.4-3 List of Containment Penetrations and System Isolation Positions (Sheet 7 of 12)

Pen NO.	GDC	System Name	Fluid	Line Size (in.)	ESF or Support System	Valve Arragmt Figure 6.2.4-1	Valve Number	Location of Valve	Type Tests	Type C Test	Length of Pipe (Note 1)	Valve Type	Valve Operator	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time	Power Source (Note 2)	Remark
P408	~~57~~56	VWS	Chilled Water	10 10 3/4	No	Sht. 43	VWS-VLV-421 VWS-MOV-403 VWS-VLV-426	In Out In	~~A~~C	~~N~~Y	- 9.0 ft -	Check Gate Check	Self Motor Self	Auto Auto Auto	None RM None	- O -	- C -	- C -	NA FAI NA	NA T NA	NA 50 NA	NA 1E NA	
P409	~~57~~56	VWS	Chilled Water	10 10 3/4	No	Sht. 43	VWS-MOV-422 VWS-MOV-407 VWS-VLV-423	In Out In	~~A~~C	~~N~~Y	- 9.0 ft -	Gate Gate Check	Motor Motor Self	Auto Auto Auto	RM RM None	O O -	O C -	C C -	FAI FAI NA	T T NA	50 50 NA	1E 1E NA	
P265	56	RMS	Containment Atmosphere	1 1 3/4	No	Sht. 44	RMS-VLV-005 RMS-MOV-003 RMS-VLV-004	In Out in	C	Y	- 9.0 ft -	Check Globe Globe	Self Motor Manual	Auto Auto Manual	None RM None	- O C	- O C	- C C	NA FAI NA	NA T NA	NA 15 NA	NA 1E NA	
P266	56	RMS	Containment Atmosphere	1 1	No	Sht. 44	RMS-MOV-001 RMS-MOV-002	In Out	C	Y	- 9.0 ft	Globe Globe	Motor Motor	Auto Auto	RM RM	O O	O O	C C	FAI FAI	T T	15 15	1E 1E	
P231	56	ICIGS	Carbon Dioxide	3/4 3/4	No	Sht. 45	IGS-AOV-002 IGS-AOV-001	In Out	C	Y	- 9.0 ft	Dia Dia	Air Air	Auto Auto	RM RM	C C	C C	C C	FC FC	T T	15 15	1E 1E	
P405R	56	LTS	Containment Atmosphere	3/4	No	Sht. 47	LTS-VLV-002 LTS-VLV-001	In Out	C	Y	- 9.0 ft	Globe Globe	Manual Manual	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P223	56	LTS	Containment Atmosphere	3/4	No	Sht. 47	- -	In Out	B	N	- -	Flange Flange	NA NA	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P216	56	LTS	Containment Atmosphere	3/4	No	Sht. 46	- -	In Out	B	N	- -	Flange Flange	NA NA	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P218	56	LTS	Containment Atmosphere	3/4	No	Sht. 46	- -	In Out	B	N	- -	Flange Flange	NA NA	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P418R	56	RLS	Containment Atmosphere	1 1/2	No	Sht. 48	- -	In Out	B	N	- -	Flange Flange	NA NA	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P418L	56	RLS	Containment Atmosphere	1 1/2	No	Sht. 48	- -	In Out	B	N	- -	Flange Flange	NA NA	Manual Manual	None None	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	
P520	56	-	-	-	-	Sht. 49	-	NA	B	N	-	None	None	Manual	Manual	C	C	C	NA	NA	NA	NA	
P530	56	-	-	-	-	Sht. 49	-	NA	B	N	-	None	None	Manual	Manual	C	C	C	NA	NA	NA	NA	
P540	56	-	-	-	-	Sht. 50	-	NA	B	N	-	None	None	Manual	Manual	C	C	C	NA	NA	NA	NA	
P208	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P213	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P215	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P246	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P254	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P268	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P269L	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P275	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P285	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P301	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P406	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P407	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P419	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	
P420	-	(Spare)	-	-	-	~~-~~Sht. 52	-	-	A	N	-	-	-	-	-	-	-	-	-	-	-	-	

Table 6.2.4-3 List of Containment Penetrations and System Isolation Positions (Sheet 10 of 12)

Note 1 - The value is the length of pipe from containment to outermost isolation valve (or the maximum length that is not be exceeded in further design)

Note 2 - Inside and Outside valves are different Class-1E power source trains

Note 3 - The following is a list of abbreviations:

GDC	General Design Criteria
RG	Regulatory Guides
Dia	diaphram
B-fly	butterfly
O	open
C	close
LC	Locked closed
FC	Fail Closed
RM	Remote Manual
S/M	System Medium
T	Containment Vessel Isolation Signal (Same as ~~S signal~~ Containment Isolation Phase A)
P	Containment Vessel Isolation Signal (Same as ~~CV spray signal~~ Containment Isolation Phase B)
S	Safety Injection Signal
V	~~Containment Ventilation Isolation Signal~~Containment Purge Isolation Signal
FAI	Fail as is
RCPS	Reactor Control and Protection System signal
Self	actuated by the fluid pressure
NA	not applicable
LTS	Leak rate testing system
RLS	RCP motor oil collection system

Note 4 - The justification for not Type C testing the safety injection lines, residual heat removal lines, containment spray lines, safety injection pump suction lines, and CS/RHR pump suction lines is that these systems are closed systems outside containment designed and constructed to ASME III, Class 2 and Seismic Category I requirements, and as such they do not constitute a potential containment atmosphere leak path during or following a loss-of-coolant accident with a single active failure of a system component. Should the valves, including test connection valves or relief valves, leak slightly when closed, the fluid seal within the pipe or the closed piping system outside containment would preclude release of containment atmosphere to the environs. These penetrations will be tested periodically as part of the Containment Integrated leak Rate Test. Furthermore, inservice testing and inspection of these isolation valves and the associated piping system outside the containment is performed periodically under the inservice inspection requirements of ASME XI as described in subsection 3.9.6 and section 6.6. During normal operation, the systems are water filled, and degradation of valves or piping is readily detected. Therefore, in accordance with ANS 56.8-1994, Section 3.3.1, these valves are not required to be Type C tested. (Ref. 6.2-35)

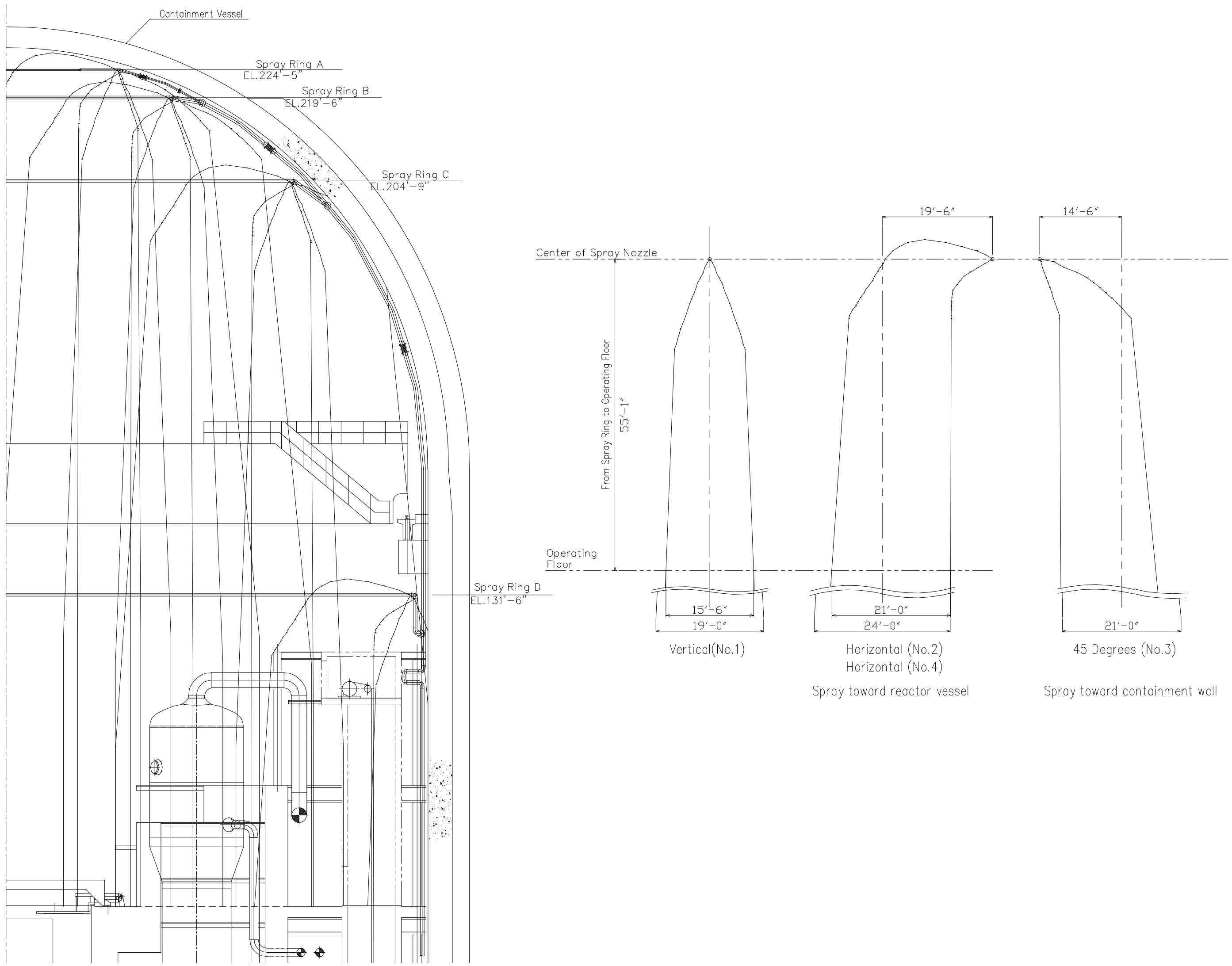

Figure 6.2.2-5 Containment Spray System Spray Ring Elevations

Reactor Coolant System

Primary Makeup Water Supply Line to Pressurizer Relief Tank

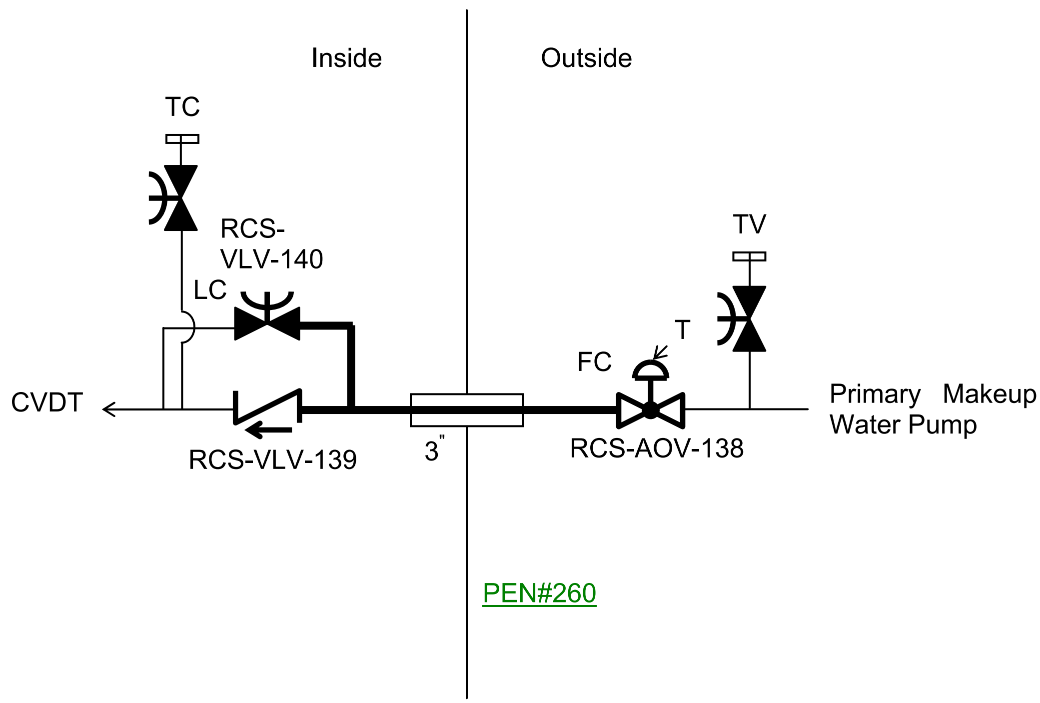

Figure 6.2.4-1 Containment Isolation Configurations (Sheet 3 of ~~51~~52)

Primary Makeup Water System

Demineralized Water Supply Line

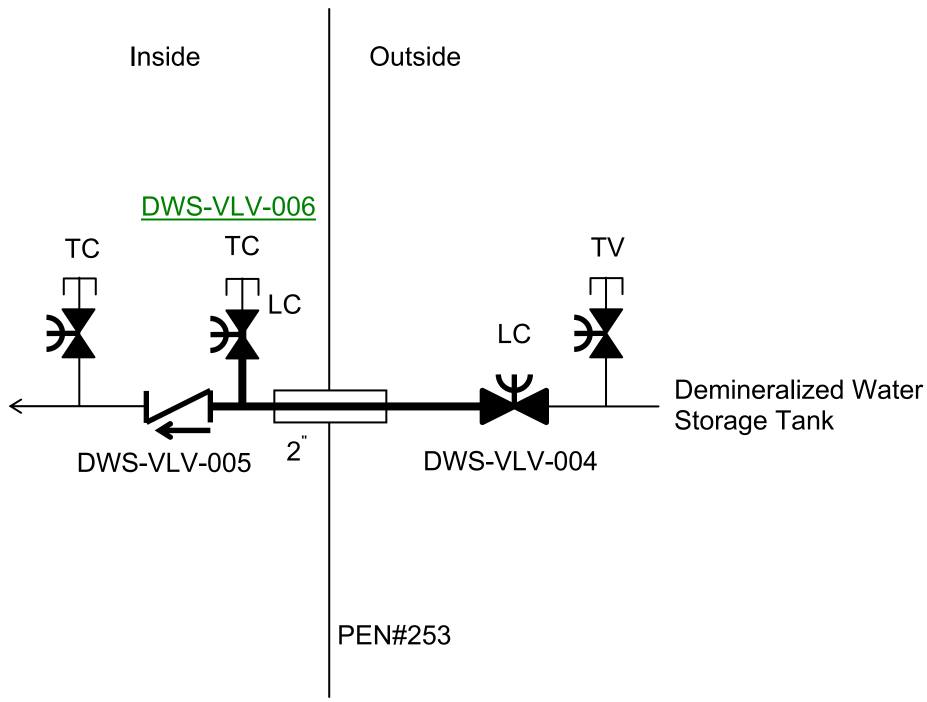

Figure 6.2.4-1 Containment Isolation Configurations (Sheet 34 of ~~51~~52)

HVAC System (Non Essential Chilled Water System)

Containment Fan Cooler Line

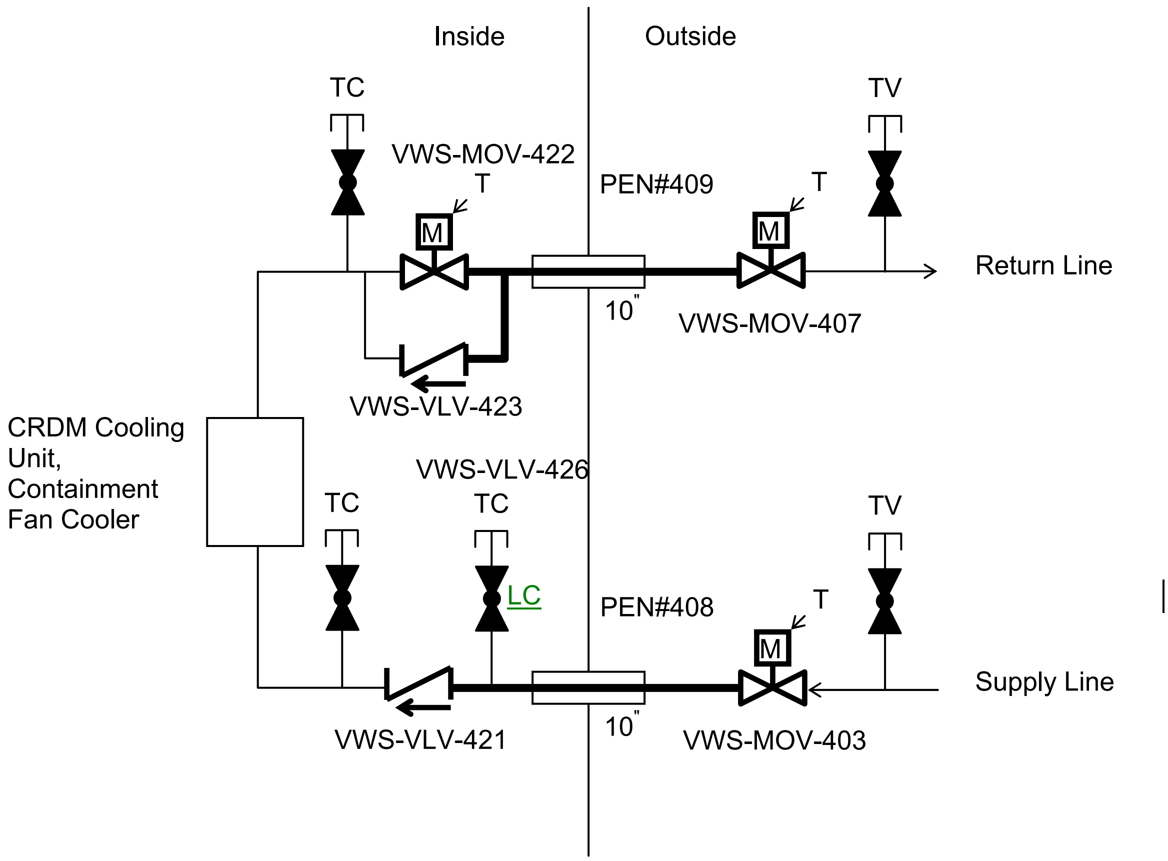

Figure 6.2.4-1 Containment Isolation Configurations (Sheet 43 of ~~51~~52)

Others

Spare Penetration

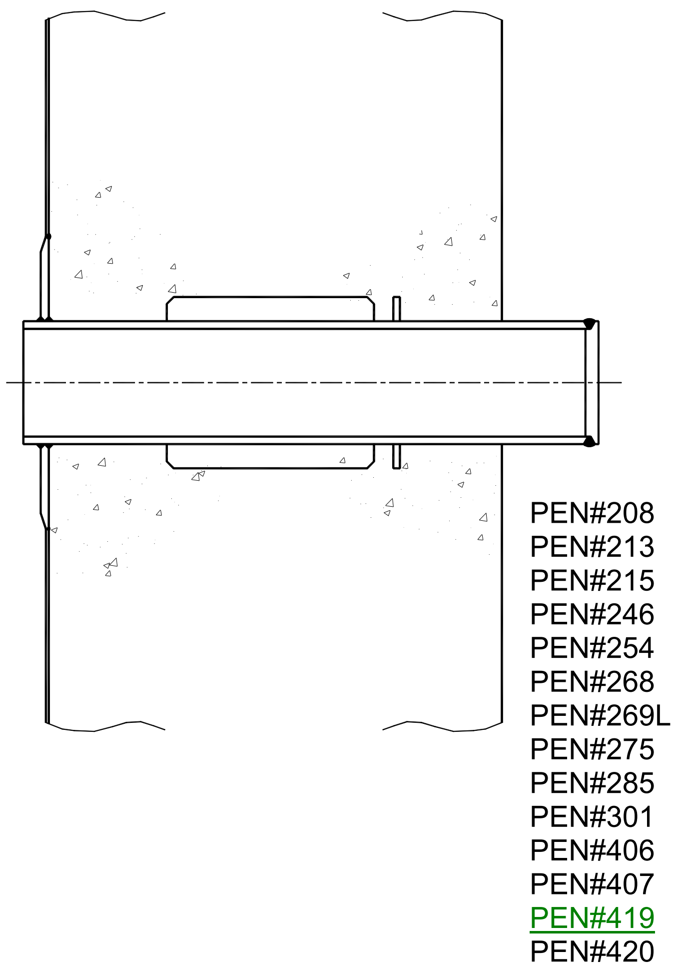

Figure 6.2.4-1 Containment Isolation Configurations (Sheet 52 of 52)

The volume of each accumulator (2,126 ft^3) includes the volume (1,342 ft^3 plus 784 ft^3) associated with both the large and small injection flow rates, respectively. Considering the total water volume (2,126 ft^3) and adding the volume of gas space and dead water volume, the required volume of a single accumulator is 3,180 ft^3 (Ref. 6.3-3).

The design temperature of the accumulator is 300°F which is consistent with the design temperature of the containment where the accumulators are located. The design pressure of the accumulator is 700 psig. This value provides margin to the normal operating pressure (i.e., nitrogen pressure) of 640 psig.

The flow rate coefficient and uncertainty of the flow damper is described in Ref. 6.3-3 and Ref. 6.3-4.

6.3.2.2.3 Refueling Water Storage Pit

The RWSP is designed to have a sufficient inventory of boric acid water for refueling and long-term core cooling during a LOCA. A minimum of 81,230 ft^3 of available water is required in the RWSP. Sufficient submerged water level is maintained to secure the minimum NPSH for the SI pumps. The RWSP capacity includes an allowance for instrument uncertainty and the amount of holdup volume loss within the containment. The capacity of the RWSP is optimized for a LOCA in order to prevent an extraordinarily large containment. Therefore, a refueling water storage auxiliary tank containing 29,410~~47,680~~ ft^3 is provided separately outside the containment to ensure that the required volume for refueling operations is met. Table 6.3-5 presents the relevant RWSP data. Detail description of structure and capacity of RWSP is provided in Subsection 6.2.2.2.

The temperature during normal operation is in a range of 70 to 120°F. The peak temperature following a LOCA is approximately 250°F.

The boric acid water in the RWSP is purified using the refueling water storage system (RWS). The RWS is shown in Figure 6.3-7 and may be cross-connected to one of two SFPCS filter and demineralizer vessels to remove the solid materials and the dissolved impurities for purification. The capacity of the purification subsystem is designed to maintain the chemistry of the spent fuel pool, the refueling cavity, the refueling water storage auxiliary tank, and the RWSP. Chapter 9, Subsection 9.1.3, discusses the SFPCS purification of the boric acid water.

6.3.2.2.4 ECC/CS Strainers

Four independent sets of strainers are provided inside the RWSP as part of the ECCS and CSS. ECC/CS strainers are provided for preventing debris from entering the safety systems, which are required to maintain the post-LOCA long-term cooling performance. ECC/CS strainers are designed to comply with RG 1.82. Strainer compliance with RG 1.82 is discussed in Subsection 6.2.2.2.6.

The RWSP is located at the lowest part of the containment in order to collect containment spray water and blowdown water by gravity. It is compartmentalized by a concrete structure against the upper containment area. Connecting pipes that drain the

6.3.2.7 Provisions for Performance Testing and Inspection

The ECCS is designed with suitable provisions that facilitate component and system performance testing. Minimum flow and full-flow test piping allow for pump testing during power operation and shutdown modes. Local instruments, test, and sample connections also support performance testing and inspection.

6.3.2.8 Manual Actions

Under LOCA conditions no operator action is required, with the exception of hot leg injection switchover. Switchover from the refueling water storage tank , in the traditional PWR, to recirculation mode is not required and the ECCS actuation signal actuates the ECCS automatically, without need for operator action.

Under normal operations, charging the accumulators (through the SI pumps) and pressurizing the accumulators with nitrogen are manual operations. Prior to the reducing reactor pressure below 1,000 psig for shutdown, the normally-open gate valve in each accumulator's discharge line is closed by remote manual operation to prevent an unintended discharge into the RCS. These valves are re-opened during startup when the reactor pressure is increased above the SI reset (un-blocking) pressure.

During safe shutdown, operator closes remotely the accumulator discharge valves by the operator's manual action before the RCS pressure decreases to the accumulator operating pressure in order to prevent the discharge of nitrogen from accumulators to the RCS. If the accumulator discharge valve could not be closed due to a single failure, operator opens remotely the accumulator nitrogen supply line isolation valve and the accumulator nitrogen discharge valve by the operator's manual action, and discharges the nitrogen in the accumulator to containment atmosphere and depressurizes the accumulator.

Operators can align any SI pump's discharge flow between the reactor vessel downcomer (normal SI flow path) and the associated reactor hot leg. Such "hot leg injection" flow prevents excessive boric acid concentration in the reactor core during long-term cooling. Hot leg injection flow is established by closing any direct vessel safety injection line isolation valve and opening the associated hot leg injection isolation valve. The valves are manually operated remotely from the MCR.

Operators manually initiate emergency letdown from the MCR. Reactor pressure is lowered by opening the safety depressurization valves, then the emergency letdown line isolation valves between the reactor hot leg ~~B~~A or D and the RWSP are opened. Borated water (at approximately 4,000 ppm boron) from the RWSP is returned to the reactor vessel through the SI pump flow, which is controlled by the associated direct vessel safety injection line isolation valve.

6.3.3 Performance Evaluation

Chapter 15 presents a complete discussion and analysis of plant anticipated operational occurrences (AOOs), transients and postulated accidents (PAs), while Chapter 19 presents a probabilistic risk assessment of more severe and even less likely accidents.

6.4.2.2.3 Isolation Dampers

MCR Air Intake Isolation Dampers:

- Two motor-operated air-tight dampers are installed in series in the outside air intake of the MCR HVAC system. These dampers are isolated in isolation mode. The two dampers are in series for single failure considerations.

MCR Toilet/Kitchen Exhaust Line Isolation Dampers:

- Two air-operated air-tight dampers are interlocked with the MCR toilet/kitchen exhaust fans and are installed at the inlet side of the MCR toilet/kitchen exhaust fans. These dampers are isolated in pressurization mode and isolation mode. The two dampers are in series for single failure considerations.

MCR Smoke Purge Line Isolation Dampers:

- Two air-operated air-tight dampers are interlocked with the MCR smoke purge fan and are installed at the inlet side of the MCR smoke purge fan. These dampers are isolated in pressurization mode and isolation mode. The two dampers are in series for single failure considerations.

The above mentioned isolation dampers are Equipment Class 3, seismic category I components.

6.4.2.2.4 Shutoff Dampers

MCR Emergency Filtration Unit Air Intake Damper:

- One motor-operated damper is installed in the duct between the outside air intake and the inlet side of each MCR emergency filtration unit. This damper sets the makeup air flow rate during pressurization mode.

MCR Emergency Filtration Unit Air Return Damper:

- One motor-operated damper is installed in the duct between the recirculation duct and the inlet side of each MCR emergency filtration unit. This damper sets the return air flow rate directed to the emergency filtration unit during pressurization mode.

The above mentioned ~~isolation~~shutoff dampers are Equipment Class 3, seismic category I components.

6.4.2.3 Leaktightness

The potential leak paths (out-leakage) of the CRE are cable, pipe, and ductwork penetrations, doors, and HVAC equipment. The extent of out-leakage (and therefore pressurization) is dependent on the sealing characteristics, and integrity, at penetrations and doors. Total system inleakage in emergency pressurization mode is equal to or less

Table 6.4-1 Main Control Room Emergency Filtration System - Equipment Specifications

Description	Specification
1. Main Control Room Emergency Filtration Units	
Auxiliaries	High efficiency prefilter, Electric heating coil, HEPA filter, Charcoal adsorber, High efficiency afterfilter
Quantity	2 (100% capacity) trains
Electric Heating Coil Capacity	18.0 kW
Charcoal Iodine Removal Efficiency (Elemental and Organic)	95% minimum
Charcoal adsorber type	Impregnated activated carbon
Charcoal adsorber weight	Maximum loading of 2.5 mg of total iodine per gram of activated carbon
Charcoal adsorber distribution	Average atmosphere residence time of 0.25 seconds per 2 inches of adsorbent bed
HEPA particulate removal efficiency	99% minimum
HEPA Filter Type	No. Designation 8 (Table FC-4110, ASME AG-1, based on 2,000 scfm(1)(2))
2. Main Control Room Emergency Filtration Unit Fans	
Quantity	2 (1 per Train)
Type	Centrifugal
Design Air Flow Rate	3,600 ft^3/min
3. Main Control Room HVAC System Isolation Dampers	
Type	Leak-tight Damper, Motor-Operated or Air-Operated
Closure Time	Less than or equal to 10 seconds

Note:

(1) Cubic foot of air per minute with a standard density.

(2) Each Main Control Room Emergency Filtration Unit has a HEPA filter assembly consisting of two of HEPA filters in parallel, for a total airflow capacity of 4000 scfm.

6.5 Fission Product Removal and Control Systems

The fission product removal systems are ESFs that remove fission products that are released from the reactor core as a result of postulated accidents and become airborne. The containment controls the leakage of fission products from the containment to ensure that the leakage fraction that may reach the environment is below limits. The US-APWR fission product removal (three systems) and control (containment) systems are as follows:

- MCR HVAC system (includes the MCR emergency filtration system)
- Annulus emergency exhaust system
- Containment spray system
- Containment

The fission product removal effects under accident conditions are shown in Table 6.5-1.

The annulus emergency exhaust system is separate and distinct from the MCR HVAC system, which is described in Section 6.4 above. The containment spray system for containment cooling is described in Subsection 6.2.2.

6.5.1 ESF Filter Systems

The annulus emergency exhaust system is one of the ESF filter systems and is designed for fission product removal and retention by filtering the air it exhausts from the following areas following accidents:

- Penetration areas
- Safeguard component areas

The penetration areas are located adjacent to the containment and include all piping and electrical penetration areas. The safeguard component areas are located adjacent to the containment and include ECCS components and CSS components that are installed outside of containment. The penetration areas and the safeguard component areas are shown in Figure 6.5-2 through 6.5-9.

The annulus emergency exhaust system is automatically initiated by the ECCS actuation signal and is initiated manually during non-ECCS actuation events (e.g., rod ejection accident or containment radiation level in excess of the normal operating range). This system establishes and maintains a negative pressure in the penetration areas and safeguard component areas relative to adjacent areas. Any airborne radioactive material in the penetration areas and safeguard component areas is directed to the annulus emergency exhaust system, avoiding an uncontrolled release to the environment.

Safeguard Component Area Supply and Exhaust Line Isolation Dampers

As shown in Figure 6.5-1, eight supply and eight exhaust line isolation dampers are normally open to provide ventilation and to maintain slightly negative pressure to the four safeguard component areas during normal operation. These isolation dampers close upon the receipt of an ECCS actuation signal. Two isolation dampers are in series for each of the four safeguard component areas for single failure considerations. Further details on the auxiliary building HVAC system are provided in Chapter 9, Subsection 9.4.3. The safeguard component area supply and exhaust line isolation dampers are Equipment Class 2, seismic category I components.

Annulus Emergency Exhaust Filtration Unit Outlet Damper

As shown in Figure 6.5-1, one ~~motor-operated~~electro hydraulic operated annulus emergency exhaust filtration unit outlet damper is installed at each fan outlet and interlocked with the annulus emergency exhaust filtration unit fan. These shutoff dampers open upon the receipt of an annulus emergency exhaust filtration unit fan run signal. The annulus emergency exhaust filtration unit outlet dampers are Equipment Class 2, seismic category I components. The annulus emergency exhaust filtration unit outlet damper is powered from Class 1E power supplies.

Safeguard Component Area Exhaust Damper

As shown in Figure 6.5-1, two safeguard component area exhaust ~~motor-operated~~electro hydraulic operated shutoff dampers are installed in parallel between the annulus emergency exhaust filtration unit fan inlet and the safeguard component area. These shutoff dampers open upon the receipt of an annulus emergency exhaust filtration unit fan run signal to maintain a negative pressure to the safeguard component areas during post-accident operation. The safeguard component area exhaust dampers are Equipment Class 2, seismic category I components. The safeguard component area exhaust dampers are powered from Class 1E power supplies.

Penetration Area Exhaust Damper

As shown in Figure 6.5-1, two penetration area exhaust ~~motor-operated~~electro hydraulic operated shutoff dampers are installed in parallel between the annulus emergency exhaust filtration unit and the penetration area exhaust header. These shutoff dampers open upon the receipt of an annulus emergency exhaust filtration unit fan run signal to maintain a negative pressure to the penetration areas during post-accident operation. The penetration area exhaust dampers are Equipment Class 2, seismic category I components. The penetration area exhaust dampers are powered from Class 1E power supplies.

Penetration Area Exhaust Backdraft Damper

As shown in Figure 6.5-1, one backdraft damper is installed on a common exhaust duct header from the A and B penetration area and another backdraft damper is installed on a common exhaust duct header from the C and D penetration area. These backdraft dampers close to prevent drawing airflow backwards through the annulus emergency

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-2 Safeguard Component Area and Penetration Area at Elevation -26'-4" – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-3 Safeguard Component Area and Penetration Area at Elevation -8’-7” – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-4 Safeguard Component Area and Penetration Area at Elevation 3’-7” – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-5 Safeguard Component Area and Penetration Area at Elevation 13’-6” – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-6 Safeguard Component Area and Penetration Area at Elevation 25'-3" – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-7 Safeguard Component Area and Penetration Area at Elevation 35'-2" – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-8 Safeguard Component Area and Penetration Area at Elevation 50’-2” – Plant View

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 6.5-9 Safeguard Component Area and Penetration Area at Elevation 76'-5" – Plant View

6.6 Inservice Inspection of Class 2 and 3 Components

Regular and periodic examinations, tests, and inspections of pressure retaining components and supports are required by 10CFR50.55a(g) (Ref. 6.6-1). This section discusses the Inservice Inspection program to address these requirements.

This section includes preservice and inservice examinations and system pressure tests. The COL Applicant is responsible for identifying the implementation milestones for ASME Section XI inservice inspection program for ASME Code Section III Class 2 and 3 systems, components (pumps and valves), piping, and supports, consistent with the requirements of 10 CFR 50.55a (g).

6.6.1 Components Subject to Examination

Chapter 3, Section 3.2, identifies the ASME Code Section III Class 2 and 3 components as corresponding quality group B and C components. Class 2 and 3 pressure-retaining components and supports subject to examination include pressure vessels, piping, pumps, valves, and their bolting. Preservice and inservice examinations, tests and inspections are performed in accordance with ASME Code Section XI (Ref. 6.6-2), including associated Mandatory Appendices, Table IWC-2500-1 for Class 2 components, and Table IWD-2500-1 for Class 3 components. The preservice inspection and ISI of threaded fasteners, in accordance with the requirements and the criteria of ASME Code, Section XI for bolting and mechanical joints used in ASME Code Class 2 systems, is described in Subsection 3.13.2.

The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before the initial fuel load. Inservice inspection of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval, subject to the limitations and modifications listed in 10 CFR 50.55a(b). In addition, the optional ASME Code cases listed in RG 1.147 may be used. The ASME Code includes requirements for system leakage tests for active components. The requirements for system leakage tests are defined in ASME Section XI, Article IWC-5220 for Class 2 pressure retaining components and ASME Section XI, Article IWD-5220 for Class 3 pressure retaining components (Ref. 6.6-2). These tests verify the pressure boundary integrity in conjunction with inservice inspection.

The preservice inspection program (non-destructive baseline examination) includes the selection of areas subject to inspection, non-destructive examination method, and the extent of preservice inspection. The inservice inspection program provides the areas subject to inspection, non-destructive examination method and extent and frequency of inspection. The inservice inspection program and inservice testing programs are submitted to the NRC. These programs comply with applicable inservice inspection and testing provisions of 10CFR50.55a(g) and (f).

Space is provided to handle and store insulation, structural members, shielding, and other materials related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment are installed at appropriate locations.

Space is provided in accordance with IWA-1500(d) for the performance of examinations alternative to those specified in the event that structural defects or modifications are revealed that may require alternative examinations. Space is also provided per IWA-1500(e) for necessary operations associated with repair/replacement activities.

6.6.3 Examination Techniques and Procedures

Surface, volumetric, and visual examinations are required for ASME Code Class 2 pressure retaining components and their welded attachments per Table IWC-2500-1. Visual examinations only are required for ASME Code Class 3 pressure retaining components and their welded attachments per Table IWD-2500-1.

A wide range of non-destructive tests for volumetric and surface material defects continue to be developed. Ultrasonic techniques are generally employed where volumetric examination is required, and either liquid penetrant or magnetic particle techniques are employed where surface examination is required. Visual examinations are conducted in accordance with the requirements of Subarticle IWA-2210 of ASME Section XI. This approach takes advantage of the most up-to-date information and experience available, as well as ensuring an inspection program acceptable to the operating organization. Qualification of the ultrasonic inspection equipment, personnel, and procedures is in compliance with Appendix VII and Appendix VIII of the ASME Code Section XI (Ref. 6.6-2). The liquid penetrant method, eddy current, ~~ultrasonic,~~ or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations.

Sufficient radial clearances are provided around pipe or component welds requiring volumetric or surface examination for inservice inspection.

Code Cases accepted for use by the NRC or appearing in RG 1.147 (Ref. 6.6-3), "Inservice Inspection Code Case Acceptability", ASME Section XI (Ref. 6.6-2), Division 1, may be applied.

6.6.4 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWC-2400 for ASME Code Class 2 components and IWD-2400 for ASME Code Class 3 components. The interval may be reduced or extended by as much as one year in accordance with ASME Code Subarticle IWA-2430 so that inspections may coincide with plant outages. Inservice examinations and system pressure tests for Class 2 and 3 components may be performed during system operation or during plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

6.6.5 Examination Categories and Requirements

Preservice examinations of ASME Code Class 2 components are performed in accordance with ASME Code Section XI (Ref. 6.6-2), Subarticle IWC-2200. Preservice examinations of Class 3 components are performed in accordance with ASME Code Section XI (Ref. 6.6-2), Subarticle IWD-2200. Similarly, Class 2 examination categories meet the requirements of Table IWC-2500-1 and Class 3 examination categories meet the requirements of Table IWD-2500-1. If alternate examination methods are used, the examination method will meet the requirements of Subarticle IWA-2240 as modified by 10CFR50.

Examination categories for ASME Code Class 2 pressure retaining components include the following:

- C-A, pressure retaining welds in pressure vessels
- C-B, pressure retaining nozzle welds in pressure vessels
- C-C, weld attachments for vessels, piping, pumps, and valves
- C-C, pressure retaining bolting greater than 2 inches in diameter
- C-F-1, pressure retaining welds in austenitic stainless steel or high alloy piping
- C-F-2, pressure retaining welds in carbon or low alloy piping
- C-G, pressure retaining welds in pumps and valves
- C-H, all pressure retaining components

Examination categories for ASME Code Class 3 pressure retaining components include the following:

- D-A, welded attachments for vessels, piping, pumps, and valves
- D-B, all pressure retaining components

6.6.6 Evaluation of Examination Results

Examination results are characterized using ASME Code Section XI (Ref. 6.6-2), Article IWA-3000 and evaluated using IWC-3000 for Class 2 components and IWD-3000 for Class 3 components. Guidelines for repair and replacement activities, if required, are according to ASME Code Section XI (Ref. 6.6-2), Article IWA-4000.

6.6.7 System Pressure Tests

System pressure testing complies with the criteria of ASME Code Section XI (Ref. 6.6-2), Article IWC-5000, for Class 2 systems, while the criteria of Article IWD-5000 apply for Class 3 systems. System leakage testing may be performed in accordance with IWC-5220 and IWD-5220 for Class 2 and 3 pressure retaining components (Categories C-H and D-B, refer to Subsection 6.6.5). A system leakage test requires the segment of the system to be tested to be inservice at system pressure performing its normal operating function, or at the system pressure developed during a test conducted to verify system operability. In lieu of a system leakage test, a hydrostatic test may be used in accordance with IWC-5230 for Class 2 pressure retaining components or IWD-5230 for Class 3 pressure retaining components.

6.6.8 Augmented ISI to Protect against Postulated Piping Failures

An augmented ISI program is required for high-energy fluid system piping between containment isolation valves or—where no isolation valve is used inside containment—between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. The ISI program contains information addressing areas subject to inspection, method of inspection, and extent and frequency of inspection in accordance with the requirements of Article IWC-2000 for Examination ~~Category C-f~~ Categories C-F-1 and C-F-2 welds. The inservice examination completed during each inspection interval is a 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping. The access provisions incorporated into the design of the US-APWR provide access for personnel and equipment to inspect the affected welds. The program covers the high-energy fluid systems described in Chapter 3, Subsections 3.6.1 and 3.6.2. An augmented ISI program is required to ensure structural integrity of cold-worked austenitic stainless steel components (Refer to Subsection 6.1.1.1).

The COL Applicant is responsible for identifying the implementation milestone for the augmented inservice inspection program.

As noted in Subsection 6.6.2, the design and installed arrangement of US-APWR Class 2 and 3 components provide clearance adequate to conduct Code-required examinations.

6.6.9 Combined License Information

Any utility that references the US-APWR design for construction and Licensed operation is responsible for the following COL items:

COL 6.6(1) The COL Applicant is responsible for identifying the implementation milestone for ASME SectionXI inservice inspection program for ASME Code Section III Class 2 and 3 systems,components (pumps and valves), piping, and supports, consistent with the requirements of 10 CFR 50.55a (g).

COL 6.6(2) The COL Applicant is responsible for identifying the implementation milestone for the augmented inservice inspection program.

6.6.10 References

6.6-1. Inservice Inspection Requirements, Title 10, code of Federal Regulations, 10 CFR 50.55a(g), January 2007.

6.6-2. Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler & Pressure Vessel Code, Division 1, Section XI, American Society of Mechanical Engineers,2001 Edition with 2003 Addenda.

Chapter 7

US-APWR DCD Chapter 7 Rev. 2, Tracking Report Rev. 7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7-xiv	ACRONYMS AND ABBREVITIONS	UAP-HF-10313: ・Added “COM communication system”
7.1-5	3rd paragraph, 7.1.1.4.2	UAP-HF-10313: Added “Command signals from safety VDUs and Operational VDUs are transmitted to the RPS, ESFAS and SLS via the PSMS communication system (COM). COM is the interface system between the safety-related PSMS and non-safety related PCMS. It provides command priority logic between the safety VDUs and operational VDUs. Command signal path and command priority logic is described in Technical Report MUAP-07004 Figure E-1 and Section 5.1.13, respectively.”
7.1-9	2nd paragraph, 7.1.3.5	UAP-HF-10313: Added “The conformance of the interdivisional communication design in the US-APWR PSMS to the Staff Positions of DI&C-ISG-04 (Reference 7.1-30) is described in Technical Report MUAP-07004, Appendix E.”
7.1-9	3rd paragraph, 7.1.3.5	UAP-HF-10313: Deleted “credible”.
7.1-10	1st paragraph, 7.1.3.7	UAP-HF-10313: Deleted “Topical Report”.
7.1-11	3rd paragraph, 7.1.3.7	UAP-HF-10313: Deleted “Topical Report”.
7.1-11	6th paragraph, 7.1.3.7	UAP-HF-10313: Deleted “Topical Report”.
7.1-12	3rd paragraph, 7.1.3.8	UAP-HF-10313: Added “MELTAC is the only digital platform used for the safety systems of the US-APWR. All other safety I&C components are conventional analog.”
7.1-12	1st paragraph, 7.1.3.9	UAP-HF-10313: Deleted “Topical Report”.
7.1-12	1st paragraph, 7.1.3.10	UAP-HF-10313: Added “Therefore, there is no impact to channel independence, system integrity and compliance to the single failure criterion during self-testing.”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.1-12	2nd paragraph, 7.1.3.10	UAP-HF-10313: Added “For PSMS, this software memory check requires temporarily connecting each PSMS controller to the Maintenance Network. When a PSMS controller is connected to the Maintenance Network, it is considered inoperable. The functions affected by an inoperable controller are managed by plant technical specifications. PCMS controllers are permanently connected to the Maintenance Network.”
7.1-12	3rd paragraph, 7.1.3.10	UAP-HF-10313: Deleted “Topical Report”.
7.1-13	1st paragraph, 7.1.3.11	UAP-HF-10313: · Added “specifically”. · Added “sensor calibration,”. · Added the following sentence: “These manual tests also recheck the portions of the system that are self-tested, and thereby manually confirm the integrity of self-tested components and the integrity of the self diagnostic functions.”
7.1-13	2nd paragraph, 7.1.3.11	UAP-HF-10313: · Added “measurement channels,”. · Replaced “continuously checks for” with “prevents”. · Added “other RPS functions and” and “maintenance”. · Added the following description: “Maintenance Bypasses may be manually initiated from the safety VDU for each respective PSMS train. To manually initiate a Maintenance Bypass from the operational VDU, the Bypass Permissive for the train must be enabled. The Bypass Permissive is part of the PSMS. There is one Bypass Permissive for each train. Administrative controls ensure the Bypass Permissive for only one train is enabled at any time. The manual Bypass Permissive is available from soft switches on the safety VDU.”
7.1-13	3rd paragraph, 7.1.3.11	UAP-HF-10313: · Deleted “only” and “ or manual controls within PSMS cabinets”. · Added “To manually initiate a Maintenance Bypass from the operational VDU, the Bypass Permissive for the train must be enabled”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.1-13	4th paragraph, 7.1.3.11	UAP-HF-10313: · Replaced "to allow automatic removal of" with "to automatically remove ". · Deleted "or operational VDUs". · Added the following sentence: "To manually initiate an Operating Bypass from the operational VDU, the Bypass Permissive for the each train must be enabled, one train at a time."
7.1-13	5th paragraph, 7.1.3.11	UAP-HF-10313: Deleted "Some safety functions may be manually overridden at the train level by deliberate manual operator action to accommodate expected plant conditions after safety function actuation. Manual overrides are administratively controlled by plant procedures. Manual overrides cannot be initiated before the safety function actuates, therefore they can never block the safety function. Manual overrides are automatically removed when the overridden signal resets. Since manual overrides are controlled by interlocks within the safety system, they may be manually initiated from safety VDUs or operational VDUs. ".
7.1-14	6th paragraph, 7.1.3.11	UAP-HF-10313: · Added "at the component level". · Deleted "or operational VDUs". · Added the following description: "This is referred to as the Lock function in MUAP-07004 Appendix D. The Lock function can also be used to block or override safety functions at the component level." · Replaced "Bypass for" with "To Lock".

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.1-14	6th paragraph, 7.1.3.11	UAP-HF-10313: · Deleted “requires a train level permissive signal from two deliberate actions from a safety VDU”. · Added “, the Bypass Permissive for the train must be enabled”. · Replaced “permissive” with “Bypass Permissive”. · Replaced “bypasses” with “Lock”. · Deleted the following descriptions “Since administrative controls allow the train level bypass permissive for the operational VDU to be enabled from the safety VDU for only one train at a time, bypasses for multiple trains are activated from the safety VDUs.” · Added “in accordance with plant technical specifications”
7.1-14	7th paragraph, 7.1.3.11	UAP-HF-10313: · Deleted “and operational VDU”. · Added the following description: “Reset signals from the operational VDU cannot be received by the PSMS without a manual Bypass Permissive signal from the safety system. If undetected reset signals exist at the time the Bypass Permissive is manually actuated, the reset errors will be indicated to operators by ESFAS reset demand status indication for the specific functions affected. The Bypass Permissive ensures additional spurious reset signals cannot be received by the PSMS at the time an AOO or PA occurs.” · Deleted the following description: “ESFAS logic allows reset operations from operational VDUs only with a permissive signal. This permissive is automatically de-activated by ESF actuation signals and can be activated from safety VDU after ESF actuation signals reset.”
7.1-14	Last paragraph, 7.1.3.11	UAP-HF-10313: Replaced “and resets” with “resets and Bypass Permissives”.
7.1-15	3rd paragraph, 7.1.3.14	UAP-HF-10313: · Replaced “will include” with “encompass”. · Added “(s)”.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.1-15	4th paragraph, 7.1.3.14	UAP-HF-10313: Added the following description as the 4th paragraph: "The method of testing for indicating and non-indicating sensors is the same. Any operational or maintenance VDU, that obtains its digital value from the PSMS, can be used for calibration. If a sensor has no operational indications its digital value will be read using a maintenance VDU, such as the MELTAC Engineering Tool, which will be temporarily connected during CHANNEL CALIBRATION."
7.1-16	1st paragraph, 7.1.3.16	UAP-HF-10313: Replaced "Credible failures" with "Failures".
7.1-17	4th paragraph, 7.1.3.16	UAP-HF-10313: Deleted "Topical Report".
7.1-17	1st paragraph, 7.1.3.17	UAP-HF-10313: Deleted "Topical Report". Deleted "cyber security management,".
7.1-17	7.1.3.18	UAP-HF-10313: Deleted "Topical Report".
7.1-17	1st paragraph, 17.1.3.19	Correction (editorial corrections) Added "Identification shall not require frequent use of reference material."
7.1-18	7.1.3	Accompanied with review of Tier-1, Tier-2 revision content was reflected.
7.1-20	7.1.5	UAP-HF-10313: Added the item 7.1-30.
7.1-22	Table 7.1-2 Sheet 1 of 8	UAP-HF-10313: Added "7.9" to the item d.
7.1-33	Figure 7.1-1	Correction (editorial corrections) Corrected the connection configuration of CRDM Control System.
7.1-36	Figure 7.1-4	Correction (editorial corrections) Corrected expression of the "Isolation" line.
7.1-39	Figure 7.1-7	Correction (editorial corrections) Corrected expression of the "VDU" power source.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-4	2nd paragraph, 7.2.1.4.1.1	UAP-HF-10313: Deleted “Topical Report”.
7.2-5	3rd paragraph, 7.2.1.4.1.2	Correction (editorial corrections) Deleted the paragraph.
7.2-5	3rd paragraph, 7.2.1.4.1.2	Correction (editorial corrections) Added the following description: “As described in Subsection 7.2.1.4.1.1, the results of the comparison with trip setpoints in train D are sent to train A and trip signals of train A (train A partial trip) are generated as a result of 1-out-of-2 logic. Train D trip signals are generated by the same logic.”
7.2-8	Equation (2), 7.2.1.4.3.1	Correction (editorial corrections) Added “K1, K2 and K3 are coefficient constants.”
7.2-8	Equation (3), 7.2.1.4.3.1	Correction (editorial corrections) Added “K4, K5 and K6 are coefficient constants.”
7.2-9	Equation (2), 7.2.1.4.3.2	Correction (editorial corrections) Added “K7, K8 and K9 are coefficient constants.”
7.2-10	1st paragraph, 7.2.1.4.8	Correction (editorial corrections) Added “(TT)” to the first sentence.
7.2-11	2nd paragraph in the enumerated item 1 of 7.2.1.4.8	UAP-HF-10313: Deleted “Topical Report”.
7.2-12	2nd paragraph, 7.2.1.6	UAP-HF-10313: Added the following description: “Maintenance and operating bypasses may be initiated from safety VDUs. To initiate a maintenance or operating bypass from an Operational VDU, the Bypass Permissive for the train must be enabled.”
7.2-13	2nd paragraph, 7.2.1.7	UAP-HF-10313: Added the following description: “Manual reset may be initiated from safety VDUs. To initiate a manual reset from an Operational VDU, the Bypass Permissive for the train must be enabled.”
7.2-15	7.2.2.6	Correction (editorial corrections) Replaced “push buttons” with “switches”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-17	2nd paragraph, 7.2.3.1	UAP-HF-10313: Replaced “credible single failures” with “single failures”.
7.2-17	3rd paragraph, 7.2.3.1	UAP-HF-10313: Deleted “credible” from first, second and third bulleted items.
7.2-33	Figure7.2-2 (Sheet 1 of 21)	Correction (editorial correction): ・ Deleted unnecessary logic symbol explanation ・ Added the reset signal to the memory circuit.
7.2-34	Figure7.2-2 (Sheet 2 of 21)	Correction (editorial correction): ・ Added manual ECCS actuation signal to the UV and SHUNT trip circuit.
7.2-35	Figure7.2-2 (Sheet 3 of 21)	Correction (editorial correction): ・ Added the rectangular enclosure to the SOURCE RANGE HIGH VOLTAGE ENERGIZE and DE-ENERGIZE function. ・ Changed the SOURCE RANGE and POWER RANGE reactor trip reset momentary control switch appearance.(delete the unnecessary arrow line.)
7.2-36	Figure7.2-2 (Sheet 4 of 21)	Correction (editorial correction): ・ Changed the MANUAL REACTOR TRIP and MANUAL REACTOR TRIP RESET momentary control switch appearance.(delete the unnecessary arrow line.) ・ Added the rectangular enclosure to the REACTOR TRIP RESET function.
7.2-38	Figure7.2-2 (Sheet 6 of 21)	Correction (editorial correction): ・ Added the rectangular enclosure to the CVCS ISOLATION function. ・ Added the NOTE 1 to the CVCS ISOLATION. NOTE 1 is the requirement that the component should not return to the condition held prior to the advent of the actuation signal when the actuation signal is lost. ・ Changed the “MANUAL CVCS ISOLATION ACTUATION” to “MANUAL CVCS ISOLATION”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-39	Figure7.2-2 (Sheet 7 of 21)	Correction (editorial correction): ・ Added the rectangular enclosure which indicate the NOT REDUNDANT function to the MAIN FEEDWATER PUMP TRIP SIGNAL. ・ Changed the memory circuit for the automatic emergency feedwater actuation signal so that the operator can reset the latched emergency feedwater actuation signal even when the actuation signal is lost. ・ Changed the NOTE 2 in order to clarify the logic of emergency feedwater pump start with LOOP SIGNAL or ECCS ACTUATION SIGNAL.
7.2-40	Figure7.2-2 (Sheet 8 of 21)	Correction (editorial correction): ・ Added the rectangular enclosure to the “CLOSE EMERGENCY FEEDWATER ISOLATION AND CONTROL VALVES”. ・ Added the NOTE 2 to the “CLOSE EMERGENCY FEEDWATER ISOLATION AND CONTROL VALVES”. NOTE 2 is the requirement that the component should not return to the condition held prior to the advent of the actuation signal when the actuation signal is lost. ・ Delete the ADJUSTABLE TIME DELAY for High steam generator water level.
7.2-41	Figure7.2-2 (Sheet 9 of 21)	Correction (editorial correction): ・ Added the NOTE 2 to the “CLOSE MAIN STEAM LINE ISOLATION VALVES”. NOTE 2 is the requirement that the component should not return to the condition held prior to the advent of the actuation signal when the actuation signal is lost. ・ Changed the appearance of “CLOSE MAIN STEAM LINE ISOLATION VALVES”.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-42	Figure7.2-2 (Sheet 10 of 21)	Correction (editorial correction): ・ Changed the “MANUAL TURBINE BYPASS BLOCK” control switch appearance. ・ Added the rectangular enclosure to the “TRIP ALL MAIN FEEDWATER PUMPS” and “CLOSE ALL MAIN FEEDWATER ISOLATION VALVES” function. And, added the NOTE1 to these functions. NOTE 1 is the requirement that the component should not return to the condition held prior to the advent of the actuation signal when the actuation signal is lost.
7.2-43	Figure7.2-2 (Sheet 11 of 21)	Correction (editorial correction): ・ Added the NOTE 1 to the “ALL REACTOR COOLANT PUMP TRIP”. ・ Changed the “ECCS ACTUATION SIGNAL” transmitted to “REACTOR TRIP(SHEET 2)” in accordance with the change of SHEET 2. This signal is divided into the auto-signal and the manual-signal. ・ Changed the appearance of “ECCS ACTUATION SIGNAL” transmitted to SHEET 12. These signals are unified with the single-line. ・ Added the ECCS ACTUATION “FIRST OUT” ANNOUNCIATOR to “HIGH CONTAINMENT PRESSURE”.
7.2-44	Figure7.2-2 (Sheet 12 of 21)	Correction (editorial correction): ・ Added the rectangular enclosure to the “HYDROGEN IGNITER ACTUATION”.
7.2-45	Figure7.2-2 (Sheet 13 of 21)	Correction (editorial correction): ・ Changed the appearance of turbine trip function with “HIGH-HIGH STEAM GENERATOR WATER LEVEL”. And, added the NOTE2 to this function. NOTE2 is the requirement that the component should not return to the condition held prior to the advent of the actuation signal when the actuation signal is lost. ・ Added the rectangular enclosure to the “GENERATOR TRIP” function. ・ Added the rectangular enclosure which indicate the NOT REDUNDANT function to the “MAIN TURBINE STOP VALVES COLSE” signal and “LOW TURBINE EMERGENCY TRIP OIL PRESSURE” signal.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-46	Figure7.2-2 (Sheet 14 of 21)	Correction (editorial correction): ・ Added the rectangular enclosures to each actuation functions. ・ Changed the positions of the block circuit (AND symbol) and ADJUSTABLE TIME DELAY for auto actuation functions.
7.2-46	Figure7.2-2 (Sheet 14 of 21)	Change with SER for MUAP-07006-P Rev. 2, ASAI 5-9 requirement: ・ Added the “TURBINE TRIP” signal to the block logic of auto actuation functions.
7.2-47	Figure7.2-2 (Sheet 15 of 21)	Correction (editorial correction): ・ Added the rectangular enclosures to the “REDUCE TURBINE LOAD REFERENCE” function. ・ Changed the expression of NOTE1 into a singular form.
7.2-48	Figure7.2-2 (Sheet 16 of 21)	Correction (editorial correction): ・ Changed the “P1ST” to “TURBINE POWER(TURBINE INLET PRESSURE”. ・ Corrected the signal line of “TURBINE POWER(TURBINE INLET PRESSURE)” to “K_{QT}”. The signal selector output of “TURBINE POWER (TURBINE INLET PRESSURE)” is input directly to K_{QT}. ・ Changed the expression of NOTE1 into a singular form.
7.2-49	Figure7.2-2 (Sheet 17 of 21)	Correction (editorial correction): ・ Changed the appearance of “MANUAL TURBINE BYPASS CONTROL RESET SWITCH”. ・ Deleted the word “STEAM HEADER PRESSURE CONTROLLER” in PI symbol.
7.2-50	Figure7.2-2 (Sheet 18 of 21)	Correction (editorial correction): ・ Changed the appearance of input signal. ・ Deleted the word “STEAM GENERATOR PRESSURE CONTROLLER” in PI symbol. ・ Added the rectangular enclosure to the “MAIN STEAM RELIEF VALVE” and “MAIN STEAM RELIEF BLOCK VALVE”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.2-51	Figure7.2-2 (Sheet 19 of 21)	Correction (editorial correction): ・ Changed the appearance of SPRAY VALVE interlock function (valve close logic). The interlock logic is described clearly as the close signal. ・ Added the rectangular enclosure to the “SPRAY VALVE A/B”, “ALL BACKUP HEATERS” and “PROPORTIONAL HEATERS”.
7.2-52	Figure7.2-2 (Sheet 20 of 21)	Correction (editorial correction): ・ Deleted the “NO LOAD Tavg” signal to “LEVEL PROGRAM CONTROLLER”. ・ Added the rectangular enclosure to the “CHARGING FLOW CONTROL VALVE”, “ALL BACKUP HEATERS”, “ALL PRESSURIZER HEATERS” and “LETDOWN LINE NO.1/2 ISOLATION VALVE”.
7.2-53	Figure7.2-2 (Sheet 21 of 21)	Correction (editorial correction): ・ Changed the appearance of the interlock function (valve close logic) for each valves. The interlock functions are described clearly as the close signal. And, changed the NOTE1 in accordance with this change.
7.2-53	Figure7.2-2 (Sheet 21 of 21)	Correction (editorial correction): ・ Changed the control logic of “MAIN FEEDWATER BYAPASS REGULATION VALVE” from the 3-elements control to the 2-elements control. And, changed the suffix number of each gain and time-constant in accordance with this correction.
7.2-55	Figure 7.2-4	Correction (editorial correction): Added “Train C RTB-C2”.
7.2-57	Figure 7.2-6	Correction (editorial corrections) Replaced “RTBr” with “RTB”.
7.3-3	2nd paragraph, 7.3.1.2	Correction (editorial corrections) Replaced “the FMEA” with “MUAP-09020 “ Function Assignment Analysis for Safety Logic System” (Reference 7.3-11)”
7.3-4	6th paragraph, 7.3.1.2	UAP-HF-10313: Deleted “Topical Report”.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.3-4	2nd paragraph, 7.3.1.2.1	UAP-HF-10313: Added the following description: “To ensure spurious command signals from Operational VDUs cannot adversely affect multiple safety divisions, all safety components controlled by the PSMS, regardless of their position under normal operating conditions, are commanded to the correct safety position by automatic safety interlocks or automatic ESFAS actuation signals.”
7.3-7	4th paragraph, 7.3.1.5.1	Correction (editorial corrections) Deleted the bulleted item “Containment spray/residual heat removal (CS/RHR) pumps”
7.3-13	7.3.1.5.12	Correction (editorial corrections) Deleted the Subsection.
7.3-14	7.3.1.6	UAP-HF-10313: Added the following description: “Maintenance and operating bypasses may be initiated from safety VDUs. To initiate a maintenance or operating bypass from an Operational VDU, the Bypass Permissive for the train must be enabled.”
7.3-15	7.3.1.6.4	UAP-HF-10313: · Replace “system level” with “train level”. · Added “In MUAP-07004 Appendix D (e), this override is referred to as a reset.”
7.3-16	7.3.1.6.4	UAP-HF-10313: · Replace “system level” with “train level”. · Added “In MUAP-07004 Appendix D (b), this override is referred to as an operating bypass.”
7.3-17,18	7.3.1.12	Correction (editorial corrections) Added the Subsection.
7.3-19	7.3.3.1	Correction (editorial corrections) Replace “Figure 7.3-5” with “Figure 7.3-6”.
7.3-26	Table 7.3-3 Sheet 3 of 3	Correction (editorial corrections) Deleted the item 12.
7.3-28	Table 7.3-4 Sheet 2 of 2	Correction (editorial corrections) Deleted the item “Block Turbine Bypass and Cooldown Valves”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.3-30	Table 7.3-6 Sheet 1 of 2	Correction (editorial corrections) Deleted the item “Manual Bypass Control for Turbine Bypass Block Train A”, “Train B”, “Train C” and “Train D”
7.3-31	Table 7.3-6 Sheet 2 of 2	Correction (editorial corrections) Deleted the item “Manual Turbine Bypass Block Actuation Train A”, “Train B”, “Train C” and “Train D”
7.3-32 to 34	Table 7.3-7	Correction (editorial corrections) Replaced “Figure 7.3-5” with “Figure 7.3-6”.
7.3-35	Figure 7.3-1	Correction (editorial corrections) Revised the figure to be consistent with MUAP-07004.
7.3-39	Figure 7.3-5	Correction (editorial corrections) Added the figure.
7.3-40	Figure 7.3-6	Correction (editorial corrections) Replaced “Figure 7.3-5” with “Figure 7.3-6”.
7.4-3	1st paragraph, 7.4.1.4	UAP-HF-10313: Deleted “can” in the second sentence.
7.4-5	7.4.1.6.1.1	Correction (editorial corrections) Corrected left margin of the enumerated items.
7.4-5	7.4.1.6.1.1	Correction (editorial corrections) Deleted an unnecessary period from the enumerated item (7).
7.4-6	7.4.1.6.1.2	Correction (editorial corrections) Corrected left margin of the enumerated items.
7.4-6	7.4.1.6.2.1	Correction (editorial corrections) Corrected left margin of the enumerated items.
7.4-7	7.4.1.6.2.2	Correction (editorial corrections) Corrected left margin of the enumerated items.
7.4-7,8	7.4.2	Correction (editorial corrections) Corrected left margin of the enumerated items.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.4-9	7.4.2.6	UAP-HF-10313: Replaced "Analog plant instrumentation and conventional component controls are relied on for normal and safe shutdown functions" with "Analog plant instrumentation and conventional electro-mechanical component (e.g., solenoids, motor starters and switchgears) are relied on for safe shutdown functions"
7.4-14	Table 7.4-1 Sheet 3 of 6	Correction (editorial corrections) · Replaced "2 per 2 SG" with "2" in the column "Required Number" of the item "EFW Isolation Valve". · Replaced "4 per 4 SG" with "4" in the column "Actual Number" of the item "EFW Isolation Valve". · Deleted "2 electrical train assigned per SG" · Replaced "2 per pump" with "4" in the column "Actual Number" of the item "T/D-EFW Pump MS Line Steam Isolation Valve". · Deleted "per pump" in the column "Required Number" of the item "T/D-EFW Pump MS Line Steam Isolation Valve". · Added "EFW Control Valve" of the item "Components" of EFWS. · Added "No" of the item "Normal Shutdown" of EFWS. · Added "Yes" of the item "Safe Shutdown" of EFWS. · Added "2" of the item "Required Number" of EFWS. · Added "4" of the item "Actual Number" of EFWS. · Added "Table 10.4.9-3" of the item "Remarks" of EFWS. · Table 7.4-1 is redesigned to add "EFW Control Valve" of the item "Components" of EFWS.
7.4-14	Table 7.4-1 Sheet 3 of 6	Detailed engineering progress Replaced "2" with "4" in the column of the item "T/D-EFW Pump Actuation Valves".

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.4-16	Table 7.4-1 Sheet 5 of 6	Detailed engineering progress · Added "Class 1E Electrical Room In-duct heater" of the item "Components" of HVAC. · Added "Yes" in the column "Normal Shutdown" of the item "Class 1E Electrical Room In-duct heater". · Added "Yes" in the column "Safe Shutdown" of the item "Class 1E Electrical Room In-duct heater". · Added "2" in the column "Required Number" of the item "Class 1E Electrical Room In-duct heater". · Added "4" in the column "Actual Number" of the item "Class 1E Electrical Room In-duct heater". · Added "Automatic start in LOOP" in the column "Remarks" of the item "Class 1E Electrical Room In-duct heater".
7.5-1	1st paragraph, 7.5.1	Correction (editorial corrections) Added "PSMS and PCMS"
7.5-2	7.5.1.1.1	UAP-HF-10313: Replaced "credible single failures" with "single failures"
7.5-8	3rd paragraph, 7.5.1.2.1	UAP-HF-10313: Replaced "Changing PSMS controller to the enable status for the engineering tool" with "Connecting PSMS controller to the maintenance network"
7.5-10	3rd paragraph, 7.5.1.3	UAP-HF-10313: · Added "digital portion of the". · Added "the operation of". · Deleted "and digital control system portion of alarms that have self-diagnosis functions". · Added the following description "Failures in the redundant visual portions of the alarm system are easily identified by operators, since the LDP and alarm VDUs are used routinely by operators for all tasks in the MCR. Failures in the redundant audible portions of the alarm system are easily identified by operators, since distinct alarm sounds normally originate from different locations within the MCR."

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.5-15	7.5.2.1	Correction (editorial corrections) Replaced “standards, and RGs” with “standard and RG”.
7.5-16	7.5.2.2	Correction (editorial corrections) Replaced “Codes, Standards and RGs” with “code and RG”.
7.5-16	7.5.2.4	Correction (editorial corrections) Replaced “codes, standards, and RGs” with “code and NUREG”.
7.5-16	7.5.2.5	Correction (editorial corrections) Replaced “codes, standards, and RGs” with “code”.
7.5-18	Reference 7.5-20	Correction (editorial correction): Incorporation of the latest revision number of the Technical Report reference.
7.6-2	3rd paragraph, 7.6.1.1	UAP-HF-10313: · Replaced “Redundant of two” with “Redundant (two)”. · Replaced “a single failure” with “a single valve failure” · Added the following sentence: “The safety interlock prevents valve opening unless the reactor coolant pressure is less than the setpoint for RHR operation.”
7.6-3	2nd paragraph, 7.6.1.2	UAP-HF-10313: Added the following sentence: “The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.”
7.6-4	3rd paragraph, 7.6.1.3	UAP-HF-10313: Added the following sentence: “The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.”
7.6-4	2nd paragraph, 7.6.1.4	UAP-HF-10313: Added the following sentence: “The safety related interlocks precludes multiple valve misalignment due to spurious commands from Operational VDU.”
7.6-4	3rd and 4th paragraph, 7.6.1.4	UAP-HF-10313: Replaced “Pull Lock” with “Lock”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.6-5	4th paragraph, 7.6.1.5	UAP-HF-10313: Added the following sentence: ” The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.”
7.6-5	5th paragraph, 7.6.1.5	UAP-HF-10313: Added the following description: ” The bypass can be selected from the safety VDU. To select the bypass from the Operational VDU, the Bypass Permissive for the respective train must be enabled.”
7.6-5	3rd paragraph, 7.6.1.6	UAP-HF-10313: Added the following sentence: ” The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.”
7.6-5	7.6.1.7	Correction (editorial corrections) ・ Replaced “Low-pressure Letdown Line Isolation valves” with “Low-pressure Letdown Line Isolation Interlock”
7.6-6	3rd paragraph, 7.6.1.7	UAP-HF-10313: Added the following sentence: ” The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.”
7.6-6	last paragraph, 7.6.1.7	Correction (editorial corrections) Added “The signal path for these interlocks is from the local pressure transmitters to the RPS, and then SLS, which controls these MOVs via motor control centers.” to the end of the Subsection.
7.7-1	3rd paragraph, 7.7	Correction (editorial corrections) Deleted “either” and “, or by a trip of the main generator” from the second sentence.
7.7-3	8th paragraph, 7.7.1.1.2	Correction (editorial correction): ・Replaced ”over temperature interlocks,” with “over temperature interlocks”. ・ Replaced “function discussed above,” with “function discussed above”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.7-8	2nd paragraph, 7.7.1.1.9	Correction (editorial corrections) ・ Replaced “three-element feedwater control” with “two-element feedwater control” in the second bulleted item. ・ Deleted “the MFW flow signal” from the second bulleted item. ・ Added “signal” to the end of the first sentence of the second bulleted item. ・ Deleted “A separate low range feedwater flow measurement is used in the low-power SG water level control mode.”
7.7-13	1st paragraph, 7.7.1.1.12	Correction (editorial corrections) ・ Dleted “this”. ・ Replaced “safety-related” with “non-Class 1E”.
7.7-13	1st paragraph, 7.7.1.1.12	Correction (editorial corrections) Replaced “Subsection 7.3.1.5.12” with “Subsection 7.3.1.12”.
7.7-13	3rd paragraph, 7.7.1.1.12	Correction (editorial corrections) Replaced “safety-related” with “non-Class 1E”.
7.7-13	3rd paragraph, 7.7.1.1.12	Correction (editorial corrections) Added the following description: “Turbine bypass valve permissive solenoids are controlled by the PSMS to achieve high reliability of block turbine bypass function as described in Subsection 7.3.1.12.”
7.7-14	1st paragraph, 7.7.1.3	Correction (editorial correction): ・ Replaced “the PCMS,” with “the PCMS”.
7.7-19	1st paragraph, 7.7.2.3	UAP-HF-10313: Replaced “credible single random failures” with “single random failures”.
7.7-20	3rd paragraph, 7.7.2.3	UAP-HF-10313: Added the following descriptions: “Since multiple spurious commands from an operational VDU are not credible, they are not considered in the analysis of bounding AOOs. However, multiple spurious commands from an operational VDU are analyzed for their effect on the safety functions, in MUAP-07004 Appendix D.”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.7-21	7.7.2.8	UAP-HF-10313: Deleted “Topical Report”.
7.7-22	7.7.2.9	UAP-HF-10313: Added the following description: “Operational VDUs generate control commands based on two distinct operator actions, in accordance with ISG-04 Position 3.1.5.”
7.7-22	3rd paragraph, 7.7.2.10	Correction (editorial corrections) Deleted “ Cyber security control of the PCMS is described in Subsection 7.9.2.6.”.
7.7-23	7.7.5	UAP-HF-10313: Added the item 7.7-3.
7.7-25	Table 7.7-2	Correction (editorial corrections) Moved the item “Control Rod Insertion Monitoring” from Group 4 to Group 3.
7.8-1	3rd paragraph, 7.8	Correction (editorial correction): ・ Replaced “located” with “installed”
7.8-1	1st paragraph, 7.8.1	Correction (editorial correction): ・ Deleted “diverse leak detection, and”
7.8-3	2nd paragraph, 7.8.1.2	UAP-HF-10313: Deleted “Topical Report”.
7.8-5	Last paragraph, 7.8.1.2.1	Additional commitment for SER of MUAP-07006 ・ Added “all the following conditions are established:”. ・ Replaced “status” with “Status” and itemize the conditions for automatic block of DAS actuation functions. ・ Added the following item: “The turbine emergency trip oil pressure trip signal is generated when oil pressure channels exceed the trip setpoint.”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	**Description of Change**
7.8-8	1st paragraph, 7.8.2.8	Correction (editorial correction): Replaced "Cope with software CCF for PA", "Minimize the extent of software CCF", "Minimize the effects of software CCF" and "Minimize the potential for adverse interaction" with "Das is implemented to mitigate the adverse effects/impacts from digital I&C both hardware and software common cause failure (CCF). It is not to minimize the potential or extent of software CCF."
7.8-22	Figure 7.8-2	Additional commitment for SER of MUAP-07006 Added the signal "Turbine emergency trip oil pressure trip signal".
7.9-1	1st and 2nd paragraph, 7.9	UAP-HF-10313: Deleted "Topical Report".
7.9-1	4th paragraph, 7.9.1.1	UAP-HF-10313: Deleted "Topical Report".
7.9-2	1st bulleted item, 7.9.1.1.2	UAP-HF-10313: · Added "automated safety signals and". · Added the following description: "All safety components controlled by the PSMS have automated safety signals and priority logic."
7.9-2	2nd bulleted item, 7.9.1.1.2	UAP-HF-10313: · Added "automatic safety signals". · Added the following sentence: "All safety components controlled by the PSMS have automated safety signals and priority logic."
7.9-2	2nd paragraph, 7.9.1.1.2	UAP-HF-10313: Added "referred to as the COM,".
7.9-3	2nd paragraph, 7.9.1.2	UAP-HF-10313: Deleted "Topical Report".

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.9-3	3rd paragraph, 7.9.1.2	UAP-HF-10313: Added the following description: “Control commands from the safety VDU processors are interfaced to other PSMS processors in the same train via the Safety Bus and the COM. Within the COM, the safety VDU commands are combined with control commands from Operational VDUs. The priority logic in the COM ensures safety VDU commands always have priority over corresponding Operational VDU commands. In addition, this logic allows all Operational VDU commands to be blocked when the Safety VDU “Disconnect” command is selected, as shown in MUAP-07004 Figure 5.1-3.”
7.9-3	3rd paragraph, 7.9.1.3	UAP-HF-10313: Deleted “Topical Report”.
7.9-4	1st paragraph, 7.9.1.4	UAP-HF-10313: Deleted “Topical Report”.
7.9-4	1st paragraph, 7.9.1.5	UAP-HF-10313: Added the following description: “PSMS controllers are normally not connected with the maintenance network. PSMS controllers that are temporarily connected to the maintenance network are declared inoperable and the affected inoperable functions of that controller are managed by Technical Specifications. Access control for the maintenance network is described in Technical Report MUAP-07004 Section 6.4.1.”
7.9-5	3rd paragraph, 7.9.1.5	UAP-HF-10313: Deleted the following description: “, which is continuously connected to the controllers of its division. (This continuous connection is different from the temporary connection of the engineering tool described in Topical Report MUAP-07005.) This continuous connection is assured by the use of the qualified isolators, which provide physical and electrical independence. Additionally, communication independence is assured by two port memory and specific attributes of the basic software within the controllers. These design features ensure that communication with the engineering tool cannot disrupt the deterministic processing of control functions or the safety functions of the PSMS. Topical Report MUAP-07005 Section 4.1.4.2 provides a detailed description of communication independence for the engineering tool”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.9-5	4th paragraph, 7.9.1.5	UAP-HF-10313: · Replaced “The engineering tool is normally used for monitoring purposes only. However, it can also be used to change application setpoints and constants and update controller software.” with “When a MELTAC controller is temporarily connected to the maintenance network, The engineering tool can be used for monitoring MELTAC controller performance, self-testing diagnostics and functional logic execution.” · Added the following description: “The PSMS application setpoints, constants and application software are changeable only by removing the CPU module that contains the memory devices from the MELTAC controller and placing it in a dedicated re-programming chassis. When the dedicated reprogramming chassis is connected to the engineering tool, either directly or via the maintenance network, the engineering tool is used to down load changes.” · Added the following description: “The PSMS basic software is changeable only by removing and replacing the memory device that contains the software.”
7.9-5	4th paragraph, 7.9.1.5	UAP-HF-10313: Deleted “Topical Report”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.9-5	5th paragraph, 7.9.1.5	UAP-HF-10313: · Deleted “In order to update constants and setpoints or allow software installation, the controller is locally selected to write-enable, using a conventional hardware write permission switch. The details of the hardware based switch enable function are described in Topical Report MUAP-07005 Section 4.3.4.2.” · Replaced “to allow this” with “to allow controllers to be connected to the maintenance network or to allow software changes” · Replaced “for the write-enable mode” with “when a controller is connected to the maintenance network or powered down to allow CPU module removal”. · Replaced “In addition, technical specifications ensure that PSMS controllers are declared inoperable by plant operators prior to enabling the write-enable mode” with “In addition, technical specifications ensure that functions affected by powering down a PSMS controller or connecting it to the maintenance network are declared inoperable in accordance with Technical Specifications.”
7.9-6	7.9.2.2	UAP-HF-10313: Deleted “Topical Report”.
7.9-7	7.9.2.3.1	UAP-HF-10313: Deleted “Topical Report”.
7.9-8	7.9.2.3.4	UAP-HF-10313: Deleted “Topical Report”.
7.9-8	7.9.2.3.6	UAP-HF-10313: Deleted “Topical Report”.
7.9-8	7.9.2.3.7	UAP-HF-10313: Deleted “Topical Report”.
7.9-9	7.9.2.4	UAP-HF-10313: Deleted “Topical Report”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.9-9	7.9.2.5	UAP-HF-10313: Security-Related Information – Withheld Under 10 CFR 2.390 .
7.9-9	2nd paragraph, 7.9.2.6	Correction (editorial correction): Deleted "The PSMS and PCMS (including the unit management computer), and all computers connected to the station bus are controlled within the plant's cyber security program which meets the requirements of Nuclear Energy Institute (NEI) 04-04 (Reference 7.9-11). "
7.9-9	3rd paragraph, 7.9.2.6	UAP-HF-10313: Deleted "Refer to Technical Report MUAP-08003 "US-APWR Cyber Security Program" (Reference 7.9-12) for details."
7.9-10	4th paragraph, 7.9.2.6	Correction (editorial correction): Deleted the paragraph.
7.9-10	Last paragraph 7.9.2.6	UAP-HF-10313: Added "The COL applicant is to provide a description of cyber security provisions."
7.9-11	7.9.2.9	UAP-HF-10313: Deleted "Topical Report".
7.9-11	7.9.2.11	UAP-HF-10313: Deleted "Topical Report".

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
7.9-12	7.9.2.13	UAP-HF-10313: Deleted “Topical Report”.
7.9-12	7.9.3	UAP-HF-10313: Deleted “Topical Report”.
7.9-12	7.9.4	Correction (editorial correction): Deleted “No additional information is required to be provided by a COL applicant in connection with this section.”.
7.9-12	7.9.4	UAP-HF-10313: Replaced “*Deleted*” with “*The COL applicant is to provide a description of cyber security provisions.*”
7.9-13	7.9.5	Correction (editorial correction): · Deleted the item 7.9-11
7.9-13	7.9.5	UAP-HF-10313: · Deleted the item 7.9-12

ACRONYMS AND ABBREVIATIONS

ac	alternating current
ACC	accumulator
ALR	automatic load regulator
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
APWR	advanced pressurized water reactor
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
AVR	auto voltage regulator
BAT	boric acid tank
BISI	bypassed and inoperable status indication
BOP	balance of plant
BTP	branch technical position
CCF	common cause failure
CCW	component cooling water
CCWS	component cooling water system
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CHP	charging pump
COL	Combined License
COM	communication system
CPU	central processing unit
CRDM	control rod drive mechanism
CSA	channel statistical accuracy
CS	containment spray
CS/RHR	containment spray/residual heat removal
CSS	containment spray system
C/V	containment vessel
CVCS	chemical and volume control system
DAAC	diverse automatic actuation cabinet
DAS	diverse actuation system
dc	direct current
DCD	Design Control Document
DCS	data communication system
DHP	diverse HSI panel
DNB	departure from nucleate boiling
E/O	electrical to optical (or optical to electrical)

7.1.1.4.2 Safety Grade HSI

The safety VDU processors manage the displays on the safety VDUs located on the OC and the RSC. They receive process parameter information from the RPS, actuation status information from the RPS and ESFAS, and component status information from the SLS. The safety VDU processors also receive operator commands such as screen navigation and soft control from the safety VDUs.

The safety VDUs are located on the OC and RSC and provide access to information and controls for safety systems.

Command signals from safety VDUs and Operational VDUs are transmitted to the RPS, ESFAS and SLS via the PSMS communication system (COM). COM is the interface system between the safety-related PSMS and non-safety related PCMS. It provides command priority logic between the safety VDUs and operational VDUs. Command signal path and command priority logic is described in Technical Report MUAP-07004 Figure E-1 and Section 5.1.13, respectively.

7.1.1.5 Information Systems Important to Safety

7.1.1.5.1 Post Accident Monitoring

The purpose of displaying PAM parameters is to assist MCR personnel in evaluating the safety status of the plant. In accordance with RG 1.97 (Reference 7.1-6), PAM Type A, B, and C variables have redundant instrumentation and can be displayed on at least two redundant safety VDUs. Type A and B parameters are continuously displayed on the LDP and are continuously available on a safety VDU, or can be retrieved immediately.

7.1.1.5.2 Bypassed and Inoperable Status Indication

If any safety function is bypassed or inoperable at the train level, this is continuously indicated on the LDP. Other bypassed or inoperable conditions that do not result in inoperability of safety functions, at the train level, are displayed on operational VDUs but not on the LDP.

7.1.1.5.3 Plant Alarms

The alarm system provides all information necessary for detecting abnormal plant conditions. The alarm system enhances the operators ability to recognize fault conditions even when the number of faults, or their severity, are increasing. Information for all alarms is displayed on the alarm VDU, LDP, and the operational VDU. LDP alarms are located in the fixed area of the LDP.

7.1.1.5.4 Safety Parameter Display System

The safety parameter display system (SPDS) provides a display of plant parameters from which the status of plant safety system operation may be assessed. The SPDS is displayed on operational VDUs located in the MCR, TSC, and EOF. The primary function of the SPDS is to aid MCR operating personnel to make quick assessments of optic data communications cables. Functional independence between controllers is maintained through communication processors that are separate from function processors, and through logic that (1) ensures prioritization of safety functions over non-safety functions and (2) does not rely on signals from outside its own train to perform the safety function within the train.

For more detailed discussion on the methods used to ensure independence between I&C systems in different safety trains and between I&C systems in safety and non-safety systems refer to described in MUAP-07004 Appendix A.5.6 and Appendix B.5.6.

Cabinets for each train of the PSMS are located in a separate plant equipment room fire area. These fire areas are separate from the fire areas where non-safety systems are located and separate from the fire areas of the MCR and the RSR. To ensure electrical independence, fiber optic cables or qualified isolators are used to interface all signals between plant equipment room fire areas. Electrical independence is also maintained between PSMS divisions and between the PSMS and non-safety systems within the MCR and the RSR.

In addition to these plant equipment room fire areas, electrical independence and physical separation are also maintained between divisions for instrumentation inputs and plant component control outputs interfaced with PSMS cabinets. The independence between the PSMS and PCMS for shared sensors is discussed in Subsection 7.1.3.16.

7.1.3.5 Isolation

Physical separation and electrical isolation are provided between the PSMS redundant trains and between the PSMS and non-safety systems, including the PCMS. Isolation devices are incorporated into conventional interfaces, data links, and communication networks that connect redundant trains, or carry signals to or from non-safety systems. The isolation devices ensure that credible faults, such as short circuits, open circuits, or the application of credible fault voltage do not propagate between systems. Chapter 8, Subsection 8.3.1.1.11 describes conformance to RG 1.204 (Reference 7.1-24). This conformance bounds the credible electrical surges and faults that are considered for electrical isolation.

In addition, for digital interfaces communications isolation is provided to ensure functional independence between systems. Communication isolation includes communication buffers, which provide separation between communication processing, functional processing, and functional logic, which ensures prioritization of all safety functions. The conformance of the interdivisional communication design in the US-APWR PSMS to the Staff Positions of DI&C-ISG-04 (Reference 7.1-30) is described in Technical Report MUAP-07004, Appendix E.

The isolation devices provide assurance that ~~credible~~ single failures in non-safety systems will not degrade the performance of safety systems, specifically in instances where protection signals are used by non-safety systems and non-safety signals are used by safety systems. For signals interfaced between redundant trains, the isolation devices provide assurance that failures in one train will not degrade the performance of other trains. The electrical, physical and functional isolation is also discussed in other trains. The electrical, physical and functional isolation is also discussed in Subsection 7.1.3.4. The isolation from a design basis event of safety system based on the equipment qualification is discussed in Subsection 7.1.3.7. The PSMS digital components are located in a mild environment that is not impacted by any design basis event.

7.1.3.6 Integrity of Software

The design principles listed below are used for the software design of all digital safety and non-safety systems using the MELTAC platform. These principles assure simplicity and enable high efficiency in the design.

- A structured and modular architecture is applied.
- Basic software and application software are separated.
- Early detection of failures is facilitated by the self-diagnosis functions of the digital system.
- Basic software is implemented in a high level programming language. All functions execute with cyclical single task processing and no interrupts.
- Basic software performs only the minimal necessary functions, such as initialization, periodic execution of required functions, error handling, etc.
- Application software is described in a graphically symbolized manner, using the problem oriented language (POL), so that functions can be easily understood.

For the non-safety I&C systems, efficiency and reliability of design, production, testing, and maintenance are achieved by using the same basic software and the design tools for the application software as is used for the safety systems.

The V&V program executed for safety systems conforms to all regulatory requirements for high integrity software.

7.1.3.7 Qualification and Equipment Protection

The PSMS is qualified for worst-case environmental and seismic requirements for the place of its installation. The PSMS qualification envelopes the seismic and environmental boundary conditions for these locations are described in Sections 3.10 and 3.11. The ~~Topical Report~~ MUAP-07005 describes equipment qualification testing for the MELTAC platform. The environmental condition of the US-APWR is described in “US-APWR Equipment Environmental Qualification Program” MUAP-08015 (Reference 7.1-25).

Conducted and/or radiated electromagnetic interference (EMI) and radio frequency interference (RFI) induced by actuation of large equipment, lightning, or radio frequency emission could degrade performance of I&C systems and compromise safety. Therefore, adequate EMI/RFI protection is designed into I&C systems components, including controllers, I/O devices, and power supply circuits. In addition, optical fiber is used for data communication, which provides electrical independence and protection against EMI/RFI.

The PSMS is qualified for EMI/RFI compatibility, as demonstrated through type testing. The EMI/RFI test levels for the PSMS are intended to envelope the possible field strength necessary for expected locations. The US-APWR is designed and operated consistent with any restrictions identified in the EMI/RFI qualification report, such as wireless communication exclusion zones, and open cabinet door conditions. This feature is described in ~~Topical Report~~ MUAP-07005 Section 5.3. The susceptibility envelope defined in RG 1.180 (7.1-26) is applicable to the US-APWR and therefore applicable to the PSMS, because the US-APWR does not include any extraordinary conducted or radiated emissions sources that are not included in operating nuclear power plants today.

Safety-related I&C components, such as sensors, detectors, cables, and connectors are designed to be qualified for operation in both normal and abnormal environments, and to meet seismic requirements of the site in which they are located. Qualification is assured per applicable industry standards (IEEE Std 323-2003 [Reference 7.1-9]) and regulatory requirements (RG 1.89 [Reference 7.1-10]), as described in Section 3.11.

Instrument sensing lines are specified to be protected in compliance with RG 1.151 (Reference 7.1-11) which endorses ISA-S67-02, including freeze protection.

For details regarding PSMS qualification testing, refer to ~~Topical Report~~ MUAP-07004 Sections 5.2.1 through 5.2.5. Refer to Chapter 3 for identification of the US-APWR qualification conditions.

The inspections, tests, analyses, and acceptance criteria (ITAAC) encompass the qualification of plant instrumentation, such as transmitters and nuclear instrumentation.

7.1.3.8 Unified Architecture

A unified architecture is applied to the design of integrated digital I&C systems of the US-APWR. The unified architecture provides a high quality and reliable platform for both the safety systems and non-safety systems, which simplifies communication between these systems. Maintenance resources are standardized for every system thereby reducing human error. An integrated digital technology is also used for VDU based operation.

Specification of the hardware modules, such as central processing units (CPUs) and I/O modules, used for each subsystem is basically the same throughout the I&C architecture, except for some specific application modules (e.g., rod position interface). This approach allows the total number of required spare parts to be minimized. The configuration of the basic software, POL, and engineering tools for specification of the application software, is the same in all digital I&C subsystems using the MELTAC platform. Maintenance procedures and tools (i.e., the engineering tool) are standardized for all subsystems; therefore, the training effort for the maintenance staff and potential for human error are minimized.

MELTAC is the only digital platform used for the safety systems of the US-APWR. All other safety I&C components are conventional analog.

7.1.3.9 Redundancy Within Divisions and Systems

To prevent disturbance of the plant caused by a failure of the safety or non-safety I&C systems, a redundant configuration is applied. Failed components, including CPU, I/O modules, and communication modules, are detected by self-diagnostic features. Redundant components are arranged in configurations for continuous parallel operation or standby operation, as described in ~~Topical Report~~ MUAP-07005 Section 4.1.1.1. For standby operation, self-diagnostic features automatically switch to redundant stand-by components in case of failure. The redundant fail-over components continue uninterrupted control without causing any disturbance to the plant.

Ac power for the I&C system is supplied from two different sources. The configuration of the power supplies within each I&C subsystem ensures no loss of function due to a single failure of the electric power source.

7.1.3.10 Self-Diagnosis Function

The integrity of digital I&C components is continuously checked by their self-diagnosis features. These self-diagnostic features result in early detection of failures, and allow on-line repair that improves system availability. Information about detected failures is gathered through networks and provided to maintenance staff in a comprehensive manner. In addition, the self-diagnostic features control redundant controller configuration, to maintain all system functions, even in the presence of failures. The self-diagnosis is always working in the digital control system but does not affect system operation. Therefore, there is no impact to channel independence, system integrity and compliance to the single failure criterion during self-testing.

Continuous self-diagnostic features allow elimination of most of the manual surveillance testing required for technical specification compliance. Manual testing and manual calibration verification are specifically provided for functions with no self-diagnostics. The integrity of the self-diagnosis is confirmed by a periodic manually initiated software memory check, which includes the software memory which is used for self-diagnosis. For PSMS, this software memory check requires temporarily connecting each PSMS controller to the Maintenance Network. When a PSMS controller is connected to the Maintenance Network, it is considered inoperable. The functions affected by an inoperable controller are managed by plant technical specifications. PCMS controllers are permanently connected to the Maintenance Network.

Also, when I/O is checked by manual sensor calibration and output actuation of plant components, the digital components which are self-tested are also re-checked. This provides manual confirmation for the integrity of all digital functions. The coverage of self-diagnosis and manual test is described in ~~Topical Report~~ MUAP-07004 Sections 4.3 and 4.4. ~~Topical Report~~ MUAP-07005 Section 4.1.5.1 describes self-diagnosis. The self-testing is provided for MELTAC components of PSMS, with the exception of the conventional circuits within the I/O and PIF modules, and the touch screens of the safety VDU.

7.1.3.11 Manual Testing, Bypasses, Overrides and Resets

Manual test features are specifically provided to allow periodic testing of all functions that are not automatically tested through self-diagnostics. This includes primarily sensor calibration, manual initiation functions and final actuation of plant components. These manual tests also recheck the portions of the system that are self-tested, and thereby manually confirm the integrity of self-tested components and the integrity of the self diagnostic functions. All manual tests may be conducted on-line without full system actuation and without plant disturbance. The test of output modules for plant components is conducted along with the test of plant components. Since the reliability of the digital I&C equipment is significantly higher than the reliability of the plant components, the periodic test frequency is determined by the reliability of the plant components, not the reliability of the digital I&C equipment.

Safety systems may be placed in a bypass operation mode to allow manual testing and maintenance while the plant is on-line. For the RPS measurement channels, automatic bypass management logic ~~continuously checks for~~ prevents multiple bypassed conditions to ensure the minimum redundancy required by the technical specifications is always maintained. For other RPS functions and the ESFAS, train level maintenance bypasses are administratively controlled. Maintenance Bypasses may be manually initiated from the safety VDU for each respective PSMS train. To manually initiate a Maintenance Bypass from the operational VDU, the Bypass Permissive for the train must be enabled. The Bypass Permissive is part of the PSMS. There is one Bypass Permissive for each train. Administrative controls ensure the Bypass Permissive for only one train is enabled at any time. The manual Bypass Permissive is available from soft switches on the safety VDU.

Indication is provided for bypassed or inoperable conditions in accordance with RG 1.47 (Reference 7.1-12). Maintenance bypasses can be manually initiated ~~only~~ within the PSMS, via safety VDUs ~~or manual controls within PSMS cabinets~~. To manually initiate a Maintenance Bypass from the operational VDU, the Bypass Permissive for the train must be enabled. During this bypass mode, a single failure in the safety system will not result in a spurious plant trip or spurious system level ESF actuation.

In addition to maintenance bypasses, automatic and manual operating bypasses are provided to bypass certain protective actions that would otherwise prevent modes of operation, such as startup and shutdown. Interlocks are provided within the PSMS ~~to allow automatic removal~~ to automatically remove ~~of~~ operating bypasses. This feature allows operating bypasses to be manually initiated from safety VDUs ~~or operational VDUs~~. To manually initiate an Operating Bypass from the operational VDU, the Bypass Permissive for the each train must be enabled, one train at a time.

~~Some safety functions may be manually overridden at the train level by deliberate manual operator action to accommodate expected plant conditions after safety function actuation. Manual overrides are administratively controlled by plant procedures. Manual overrides cannot be initiated before the safety function actuates, therefore they can never block the safety function. Manual overrides are automatically removed when the overridden signal resets. Since manual overrides are controlled by interlocks within the safety system, they may be manually initiated from safety VDUs or operational VDUs.~~

All manual and automatic demand signals for components, which are controlled by motor control centers and switchgear, may be bypassed at the component level for testing or maintenance by two deliberate manual operator actions from safety VDUs~~ or operational VDUs~~. This is referred to as the Lock function in MUAP-07004 Appendix D. The Lock function can also be used to block or override safety functions at the component level. ~~Bypass for~~ To Lock safety-related components from operational VDUs ~~requires a train level permissive signal from two deliberate actions from a safety VDU~~, the Bypass Permissive for the train must be enabled. This Bypass P~~p~~ermissive is administratively controlled so that it is enabled for only one train at a time. When the permissive is enabled for a train from the safety VDU, ~~bypasses~~ Lock may be activated from the operational VDU individually for any components within that train. ~~Since administrative controls allow the train level bypass permissive for the operational VDU to be enabled from the safety VDU for only one train at a time, bypasses for multiple trains are activated from the safety VDUs.~~ All bypasses are administratively managed by plant operators in accordance with plant technical specifications. The effect of bypasses is alarmed at the system level for each train via the bypassed and inoperable status indication (BISI).

After safety function actuation and after initiating conditions return to normal, safety functions are manually reset at the system level. These resets are available from safety VDUs~~ and operational VDU~~. Reset signals from the operational VDU cannot be received by the PSMS without a manual Bypass Permissive signal from the safety system. If undetected reset signals exist at the time the Bypass Permissive is manually actuated, the reset errors will be indicated to operators by ESFAS reset demand status indication for the specific functions affected. The Bypass Permissive ensures additional spurious reset signals cannot be received by the PSMS at the time an AOO or PA occurs. ~~ESFAS logic allows reset operations from operational VDUs only with a permissive signal. This permissive is automatically de-activated by ESF actuation signals and can be activated from safety VDU after ESF actuation signals reset.~~ Maintenance bypasses, operating bypasses, overrides, resets and Bypass Permissives~~and resets~~ are initiated separately for each safety division.

7.1.3.12 Human-System Interface

The MCR is designed to perform centralized monitoring and control of the I&C systems that are necessary for use during normal operation, AOOs, and PAs. Furthermore, the HSIS are also designed to reduce the potential for human error and to allow easy operation. In addition to the MCR, the HSI also includes the RSC, TSC, EOF, and local control stations, such as auxiliary equipment control console. Refer to Chapter 18 for a full discussion of all HSI issues.

7.1.3.13 Quality of Components and Modules

The quality of PSMS components and modules and the quality of the PSMS design process are controlled by a program that meets the requirements of American Society of Mechanical Engineers (ASME) NQA-1-1994 (Reference 7.1-13). Conformance to ASME NQA-1-1994 is described further in Section 17.5.

7.1.3.14 System Calibration, Testing and Surveillance

Testing from and including the sensors of the PSMS through to and including the actuated equipment and HSI is accomplished in a series of overlapping sequential tests and calibrations. The majority of the tests are conducted automatically, through self-diagnostics. Most remaining manual tests may be performed with the plant at full power. There are no exceptions for testing at power in PSMS.

The test frequency for manual tests is based on an uncertainty and reliability analysis, reference Subsection 7.2.2.7 and 7.2.3.5, respectively, for additional information. This analysis demonstrates the need to conduct most manual tests for PSMS equipment no more frequently than once per 30 months, which allows for fuel cycles up to 24 months plus 6 months to accommodate 25% margin for consistency with technical specification surveillance interval compliance. Therefore conducting manual tests for PSMS equipment on-line or off-line, during refueling shutdown, is at the discretion of the plant owner.

Periodic routine calibration will be performed for the field located transmitters of each safety-related instrument loop. Due to the digital design of the control platform in the US-APWR, a traditional calibration method will be performed from the sensor to the analog to digital converter. During this calibration, the digital display will provide the instrument output. As in a traditional calibration, the measured value on the display will be compared to an expected range. Calibration points ~~will include~~ encompass the trip setpoint to confirm required accuracy at the trip setpoint value(s).

The method of testing for indicating and non-indicating sensors is the same. Any operational or maintenance VDU, that obtains its digital value from the PSMS, can be used for calibration. If a sensor has no operational indications its digital value will be read using a maintenance VDU, such as the MELTAC Engineering Tool, which will be temporarily connected during CHANNEL CALIBRATION.

The PSMS meets the periodic testing requirements of IEEE Std 338-1987 (Reference 7.1-27) which is endorsed by RG 1.22 (Reference 7.1-28). The test intervals are specified in the technical specifications, Chapter 16. All periodic testing is conducted to written procedures. For more detailed discussion on this topic, refer toMUAP-07004 Sections 4.3 through 4.5, Appendix A.5.7, A.5.9, A.5.10, A.6.5 through A.6.7, and A.7.5.

Installed RTDs will be calibrated using the method defined in BTP 7-13(Reference 7.1-29). The following accuracy calibration is applicable to all safety related RTDs:

- A reference RTD is checked for acceptable accuracy and response time in controlled laboratory conditions.
- The reference RTD is installed. Loop current step response (LCRS) is checked to confirm applicability of laboratory test data.
- Measurements from installed RTDs are cross correlated to the reference RTD under known and sufficiently similar temperature and flow conditions (i.e., isothermal conditions of all RCS hot and cold legs to the extent practical).
- Calibration readout will be on digital displays, as discussed above, to ensure correct signal propagation and accuracy through the digital systems.

In addition, the LCRS is checked for installed RTDs used in the RPS or ESFAS, where response times are credited in the safety analysis. To detect response time degradation, the LCRS data is compared to the installed RTD's own historical data and to the LCRS for the reference RTD.

The accuracy and response time acceptance criteria account for expected instrument uncertainties and expected temperature and flow deviations. "As found" and "as-left" data is recorded and maintained.

7.1.3.15 Information Displays

Details on information displays are presented in Topical Report MUAP-07007, Chapter 18, and Section 7.5.

7.1.3.16 Consideration of Control System Induced Transients

~~Credible f~~Failures of the PCMS are bounded by the AOOs analyzed in the safety analysis, described in Chapter 15. These PCMS failures are described in Subsection 7.7.2.3. Chapter 8, Subsection 8.3.1.1.11 describes conformance to RG 1.204. This conformance bounds the envelope considered for PCMS EMI susceptibility. The PCMS uses the same hardware as the PSMS, which is qualified to RG 1.180. Therefore, additional lightening induced failures of the PCMS are precluded.

In some cases, it is advantageous to employ signals derived from instrumentation that are also used in the protection trains. This practice reduces the need for separate non-safety instrumentation, which would require additional penetrations into reactor pressure boundaries and introduce the need to additional maintenance in hazardous areas. For each parameter where instrumentation is shared, the PCMS receives four redundant signals from each train of the RPS. The signal selection algorithm (SSA), within the PCMS, receives input from all safety process trains but passes only the second highest operable process signal value to the control system's automation algorithms. The reactor control systems also have a modified signal selector using an average calculation process. (This average calculation for select signals in the reactor control system is different from the description in MUAP-07004 Section 4.2.5.) The SSA excludes process inputs that are failed or taken out of service for maintenance or testing.

The SSA of the PCMS ensures the PCMS does not take erroneous control actions based on a single instrument channel failure or a single RPS train failure. As such, a single failure will not cause the PCMS to take erroneous control actions that challenge the PSMS, while the PSMS is in a degraded operability state due to a failed instrument channel or failed RPS train. The SSA is designed with an augmented quality program, including software V&V.

The SSA is continuously tested as follows:

- The PCMS employs the same self-test features as the PSMS. These features are described in Section 4.1.5 of ~~Topical Report~~ MUAP-07005.

- The basic software configuration and application software configuration, within the PCMS controller, is periodically confirmed by the same manually initiated method described in Section 4.1.4.1.c of ~~Topical Report~~ MUAP-07005.

Since the SSA uses only digital values obtained from the PSMS via the unit bus, all functions of the SSA are completely covered by self-testing; no additional manual tests are required. The digital values obtained from the PSMS are confirmed during channel calibration for the safety sensors.

This SSA within the PCMS allows the RPS to have one instrument channel inoperable or bypassed at all times while still complying with General Design Criteria (GDC) 24 (Reference 7.1-14) and IEEE Std 603-1991 (Reference 7.1-15). As described in the probabilistic risk assessment (PRA) the RPS meets the plant reliability goals with only three channels in operation. Refer to US-APWR Probabilistic Risk Assessment, Technical Report MUAP-07030 (Reference 7.1-16).

The shared instrumentation signals are interfaced through fiber optic data networks. As such, an electrical fault in the PCMS cannot propagate to the protection channel. Refer to MUAP-07004 Section 4.2.5 for additional details.

7.1.3.17 Life Cycle Process

MHI applies its MELCO's safety system digital platform, MELTAC, to the PSMS and PCMS systems of the US-APWR. Full details of the life cycle process for the MELTAC basic software, including quality assurance (QA), management, development, ~~cyber security management,~~ installation, maintenance, training, operation, and the software safety plan are discussed in ~~Topical Report~~ MUAP-07005 Section 6.0. A summary of the life cycle process for the system application software, including QA, management, development, ~~cyber security management,~~ installation, maintenance, training, operation, and the software safety plan are discussed in ~~Topical Report~~ MUAP-07004 Section 6.0. A detailed description of the application software life cycle process, including BTP 7-14 (Reference 7.1-17) compliance, is provided in the Software Program Manual for the US-APWR Technical Report MUAP-07017 (Reference 7.1-18).

7.1.3.18 Quality Assurance Program

The overall quality assurance program (QAP) for the US-APWR I&C systems is described in Chapter 17. The specific QAP for the MELTAC platform is described in ~~Topical Report~~ MUAP-07005 Section 6.0. These QAPs address all requirements of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B (Reference 7.1-19), and IEEE Std 7-4.3.2-2003 (Reference 7.1-20).

7.1.3.19 Identification

I&C equipments identification follows the guidance of RG 1.75, which endorses IEEE Std 384. The following color coding is provided on tags used for the identification of I&C system cabinets and for stand alone components, such as field instruments. Identification shall not require frequent use of reference material.

- Train A: Red with white lettering
- Train B: Green with white lettering
- Train C: Blue with white lettering
- Train D: Yellow with Black lettering
- Non-safety train: White with Black lettering

This color coding is consistent with the color coding defined in Subsection 8.3.1.1.8 identification of class 1E electrical equipment and cables.

For computer-based systems, the configuration management plan describes the identification process for software. To ensure that the required computer system hardware and software are installed in the appropriate system configuration, the system meets the following identification criteria specific to software systems:

- Firmware and software identification ensures that the correct software is installed in the correct hardware component.
- The software has a means to retrieve identification from the firmware by using software maintenance tools.
- Physical identification requirements of the digital computer system hardware are in accordance with the identification requirements in IEEE Std 603-1991.

The configuration identification management is addressed in Technical Report MUAP-07017.

7.1.4 Combined License Information

No additional information is required to be provided by a Combined License (COL) applicant in connection with this section.

7.1.5 References

7.1-1 Manual Initiation of Protective Actions, Regulatory Guide 1.62 Revision 0, October 1973.

7.1-2 Safety I&C System Description and Design Process, MUAP-07004-P Rev.3 (Proprietary) and MUAP-07004-NP Rev.3 (Non-Proprietary), September 2009.

7.1-3 Safety System Digital Platform -MELTAC-, MUAP-07005-P Rev.4 (Proprietary) and MUAP-07005-NP Rev.4 (Non-Proprietary), September 2009.

7.1-4 Defense-in-Depth and Diversity, MUAP-07006-P-A Rev.2 (Proprietary) and MUAP-07006-NP-A Rev.2 (Non-Proprietary), September 2009.

7.1-5 HSI System Description and HFE Process, MUAP-07007-P Rev.3 (Proprietary) and MUAP-07007-NP Rev.3 (Non-Proprietary), October 2009.

7.1-6 Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants, Regulatory Guide 1.97 Revision 4, June 2006.

Identification shall not require frequent use of reference material.

- Train A: Red with white lettering
- Train B: Green with white lettering
- Train C: Blue with white lettering
- Train D: Yellow with Black lettering
- Non-safety train: White with Black lettering

This color coding is consistent with the color coding defined in Subsection 8.3.1.1.8 identification of class 1E electrical equipment and cables.

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7.1-3 Safety System Digital Platform -MELTAC-, MUAP-07005-P Rev.4 (Proprietary) and MUAP-07005-NP Rev.4 (Non-Proprietary), September 2009.

7.1-4 Defense-in-Depth and Diversity, MUAP-07006-P-A Rev.2 (Proprietary) and MUAP-07006-NP-A Rev.2 (Non-Proprietary), September 2009.

7.1-5 HSI System Description and HFE Process, MUAP-07007-P Rev.3 (Proprietary) and MUAP-07007-NP Rev.3 (Non-Proprietary), October 2009.

7.1-23 IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, IEEE Std 384-1992.

7.1-24 Guideline for Lightning Protection of Nuclear Power Plants, Regulatory Guide 1.204 Revision 0, November 2005.

7.1-25 US-APWR Equipment Environmental Qualification Program, MUAP-08015 Rev.0, February 2009.

7.1-26 Guideline for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems, Regulatory Guide 1.180 Revision 1, October 2003.

7.1-27 Standard Criteria for Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems, IEEE Std 338-1987.

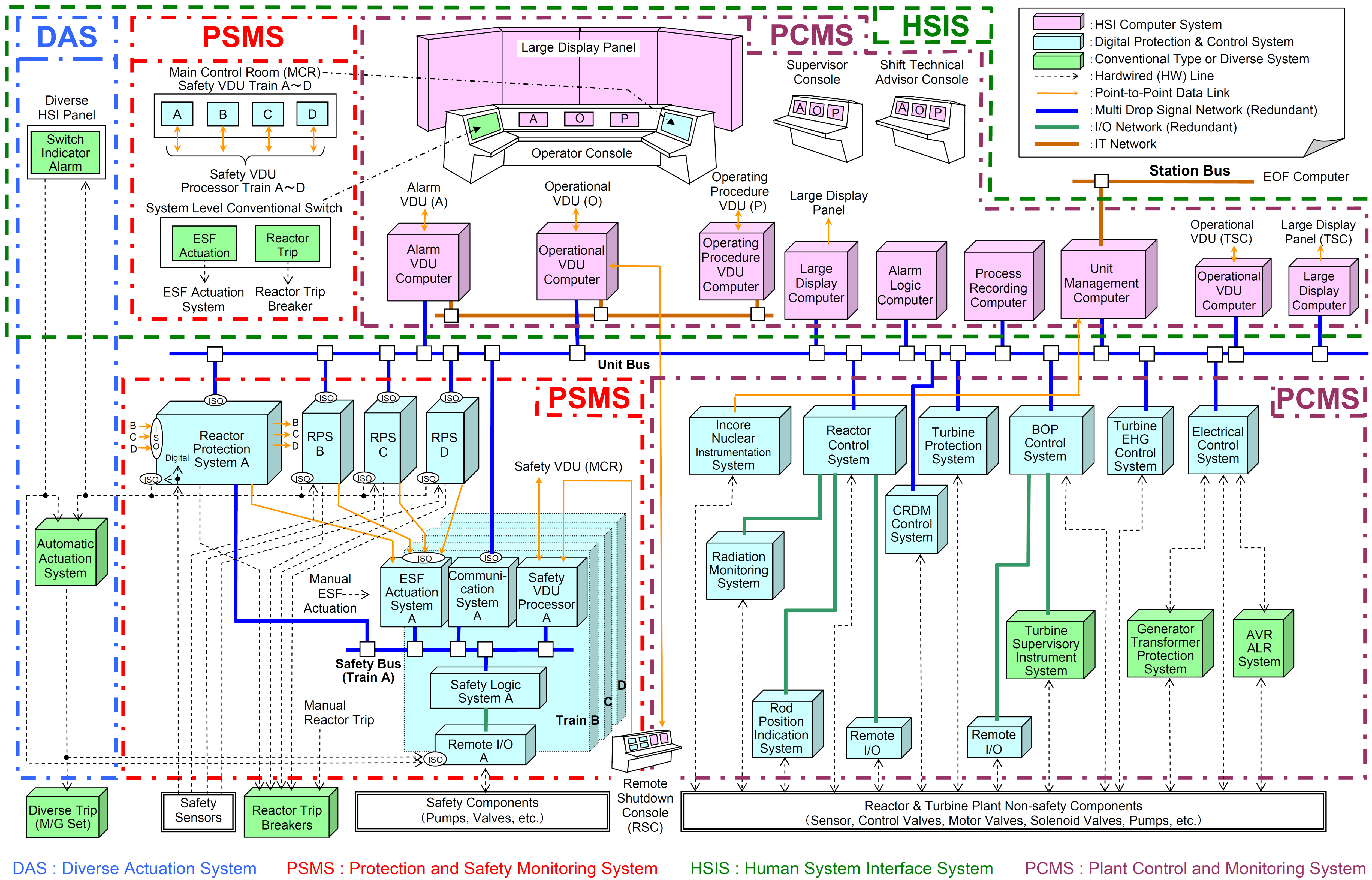
7.1-28 Periodic Testing of Protection System Actuation Functions, Regulatory Guide 1.22 Revision 0, February 1972.

7.1-29 Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors, BTP 7-13 Revision 5, March 2007.

7.1-30 Highly Integrated Control Rooms – Digital Communication Systems, DI&C-ISG-04 Revision 1, March 2009.

Table 7.1-2 Regulatory Requirements Applicability Matrix (per NUREG-0800 Standard Review Plan (SRP) Sec. 7.1 Rev. 5) (Sheet 1 of 8)

Applicable Criteria		Title	I&C System							Related Section in US-APWR DCD
			RPS	ESFAS	SLS	Safety HSI	Safety DCS	PCMS	DAS	
		1. 10 CFR 50 and 52								
a.	50.55a(a)(1)	Quality Standards for Systems Important to Safety	X	X	X	X	X			7.2 to 7.6, 7.9
b.	50.55a(h)(2)	Protection Systems (IEEE Std 603-1991 or IEEE Std 279-1971)	X	X	X	X	X			7.2 to 7.6, 7.9
c.	50.55a(h)(3)	Safety Systems (IEEE Std 603-1991)	X	X	X	X	X			7.2 to 7.6, 7.9
d.	50.34(f)(2)(v) [I.D.3]	Bypass and Inoperable Status Indication	X	X	X	X	X	X		7.2, 7.3, 7.5, 7.6, 7.9
e.	50.34(f)(2)(xi) [II.D.3]	Direct Indication of Relief and Safety Valve Position			X		X	X		7.5
f.	50.34(f)(2)(xii) [II.E.1.2]	Auxiliary Feedwater System Automatic Initiation and Flow Indication	X	X	X	X	X			7.3, 7.5
g.	50.34(f)(2)(xvii) [II.F.1]	Accident Monitoring Instrumentation	X		X	X	X	X		7.5
h.	50.34(f)(2)(xviii) [II.F.2]	Instrumentation for the Detection of Inadequate Core Cooling	X			X	X			7.5
i.	50.34(f)(2)(xiv) [II.E.4.2]	Containment Isolation Systems	X	X	X	X	X			7.3
j.	50.34(f)(2)(xix) [II.F.3]	Instruments for Monitoring Plant Conditions Following Core Damage	X			X	X			7.5
k.	50.34(f)(2)(xx) [II.G.1]	Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves	X		X	X	X			7.4, 7.5
l.	50.34(f)(2)(xxii) [II.K.2.9]	Failure Mode and Effect Analysis of Integrated Control System								N/A to US-APWR
m.	50.34(f)(2)(xxiii) [II.K.2.10]	Anticipatory Trip on Loss of Main Feedwater or Turbine Trip								N/A to US-APWR
n.	50.34(f)(2)(xxiv) [II.K.3.23]	Central Reactor Vessel Water Level Recording								N/A to US-APWR



:HSI Computer System
:Digital Protection & Control System
:Conventional Type or Diverse System
:Hardwired (HW) Line
:Point-to-Point Data Link
:Multi Drop Signal Network (Redundant)
:I/O Network (Redundant)
:IT Network

DAS : Diverse Actuation System
PSMS : Protection and Safety Monitoring System
HSIS : Human System Interface System
PCMS : Plant Control and Monitoring System

Figure 7.1-1 US-APWR I&C Overall Architecture

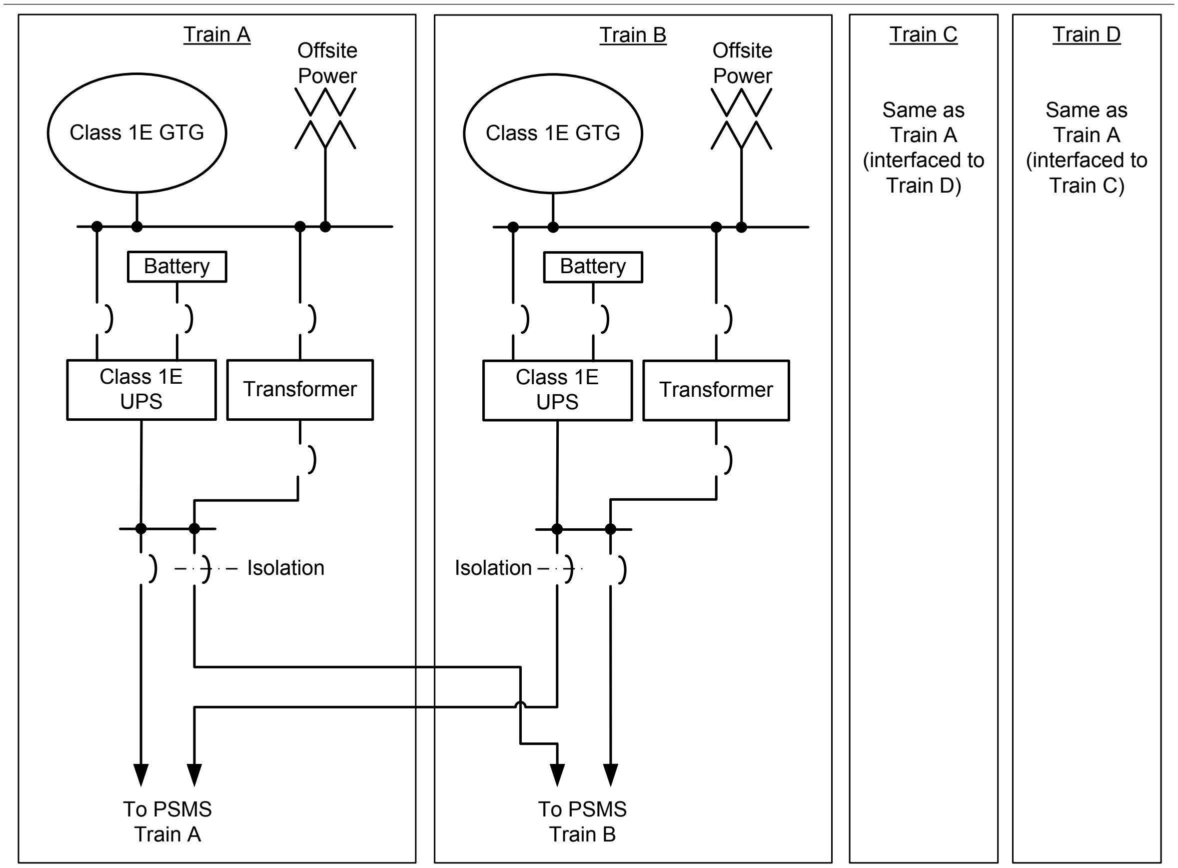


Figure 7.1-4 Class 1E UPS for PSMS

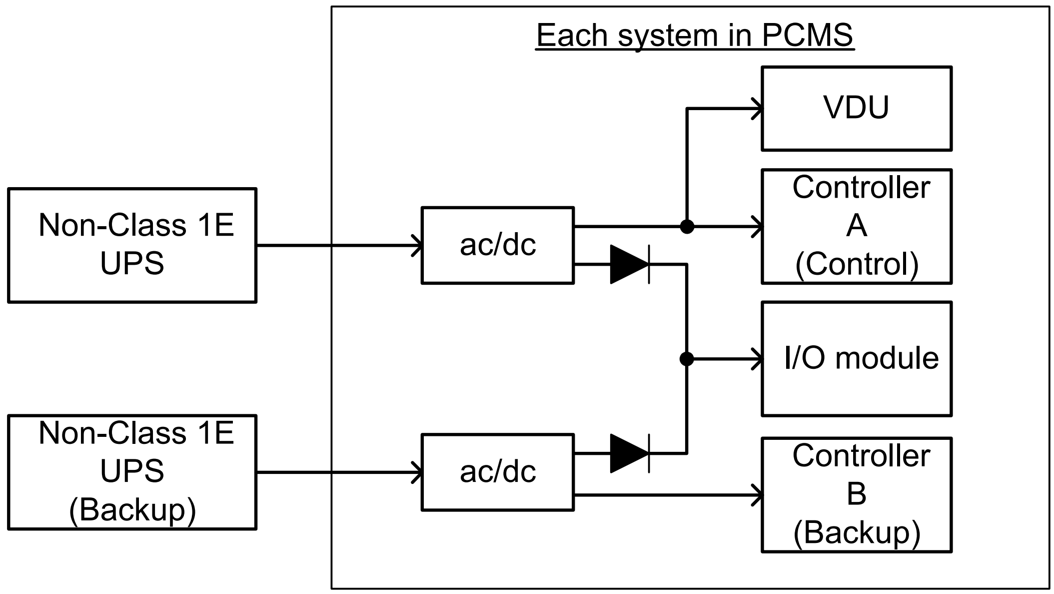


Figure 7.1-7 Electrical Power Source for PCMS

- ECCS actuation signal
- Manual RT actuation signal
- Turbine trip signal

7.2.1.4 Reactor Trip Initiating Signals

The following subsection describes the RT initiating signals that are grouped according to their protection function. Train level manual controls are identified in Table 7.2-6 for conventional switches and in Table 7.2-7 for software switches.

Pre-trip alarms and non-safety interlocks are initiated below the RT setpoints to provide audible and visible indication of the approach to a trip condition.

7.2.1.4.1 Nuclear Startup Protection Trips

7.2.1.4.1.1 High Source Range Neutron Flux

A high source range neutron flux trip provides protection during reactor startup and plant shutdown. An operating bypass may be manually initiated when the neutron flux is above the P-6 setpoint value (intermediate range), which will also de-energize the high voltage power supply to the source range neutron flux detector. This trip is automatically bypassed by the power range neutron flux interlock P-10. The bypass automatically resets to reactivate the trip function when the intermediate range neutron flux decreases below the P-6 reset point.

Due to the limited duration of reactor startup and shutdown, there are only two source range instrument channels, trains A and D. The train A source range neutron flux detector interfaces to RPS trains A and B. The train D source range neutron flux detector interfaces to RPS trains C and D. Interfaces to RPS trains B and C are via isolated digital data links described in ~~Topical Report~~ MUAP-07005 (Reference 7.2-2) Section 4.3.3. Signal flow for this trip signal is shown in Figure 7.2-6.

Source range and intermediate range neutron flux signals from the train A detectors are sent to the RPS controllers of train A and B through analog circuits installed in train A nuclear instrumentation system (NIS) cabinet and compared with trip setpoints in each train controller. The signals from the train A NIS cabinet to the train B RPS cabinet are isolated in the train B RPS cabinet. In the case of train D detectors, neutron flux signals are sent to train C and D controllers. The results of the comparison with trip setpoints in train D are sent to train A and trip signals of train A (train A partial trip) are generated as a result of 1-out-of-2 logic. Train D trip signals are generated by the same logic. Results of the 1-out-of-2 logic may be bypassed by the following operating bypass signals: (1) P-6 allows source range neutron flux trip manual operating bypass, (2) P-10 allows source range neutron flux trip automatic operating bypass, and (3) P-10 allows intermediate range trip manual operating bypass.

Similar trip logic functions are processed in each train. Partial trip signal from each train is sent to each trip breaker of the corresponding train and a 2-out-of-4 voting logic is implemented.

The source range channels can be individually bypassed in each RPS train to permit channel testing. Since there are only two source range channels, the duration of this bypass is limited by the technical specifications. This operating bypass action is indicated in the MCR. Figure 7.2-2 sheet 3 shows the logic for this trip function.

7.2.1.4.1.2 High Intermediate Range Neutron Flux

This trip provides protection during reactor startup and shutdown. It can be manually bypassed if the power range channels are above 10 percent power (P-10). This operating bypass is automatically reset to reactivate the trip function when the power range channels indicate less than the reset point for P-10.

Due to the limited duration of reactor startup and shutdown, there are only two intermediate range instrument channels, trains A and D. Train A intermediate range neutron flux detector interfaces to RPS trains A and B. Train D intermediate range neutron flux detector interfaces to RPS trains C and D. Interfaces to RPS trains B and C are by isolated digital data links.

~~The intermediate range neutron flux trip is generated using the same 2-out-of-4 voting logic as for all other RPS trips. However, since the same signal is distributed to two divisions, this results in an effective 1-out-of-2 voting logic to initiate a RT, as described in Subsection 7.2.1.4.1.1.~~

As described in Subsection 7.2.1.4.1.1, the results of the comparison with trip setpoints in train D are sent to train A and trip signals of train A (train A partial trip) are generated as a result of 1-out-of-2 logic. Train D trip signals are generated by the same logic.The intermediate range channels can be individually bypassed in each RPS train to permit channel testing. Since there are only two intermediate range channels, the duration of this maintenance bypass is limited by the technical specifications. This maintenance bypass action is indicated in the MCR. Figure 7.2-2 sheet 3 shows the logic for this trip function.

7.2.1.4.1.3 High Power Range Neutron Flux (Low Setpoint)

This parameter trips the reactor when two of the four power range channels exceed the trip setpoint. This trip provides protection during startup. It can be manually bypassed when the power range channels are above 10 percent power (P-10). This operating bypass automatically resets to reactivate the trip function when power range channels indicate less than the reset point for P-10. This operating bypass action is indicated in the MCR. Figure 7.2-2 sheet 3 shows the logic for this trip function.

7.2.1.4.2 Nuclear Overpower Protection Trips

Four power range nuclear instrumentation detectors are installed vertically at the four corners of the core. Each power range neutron flux detector assembly consists of an

K1, K2 and K3 are coefficient constants.

(3) Core exit boiling protection setpoint

$$\Delta T_{SP2} = \Delta T_0 \left(K_4 - K_5 \frac{(1+T_4 s)}{(1+T_5 s)} \left(T_{avg} - T_{avg0}\right) + K_6 \left(P - P_0\right) \right)$$

Where: ΔT_{SP2} is the core exit boiling protection setpoint.

ΔT_0 is the indicated RCS ΔT at RTP.

s is the Laplace transform operator.

T_{avg} is the measured RCS average temperature.

T_{avg0} is the nominal T_{avg} at rated thermal power.

P is the measured pressurizer pressure.

P_0 is the nominal RCS operating pressure.

K4, K5 and K6 are coefficient constants.

7.2.1.4.3.2 Over Power Delta T

This trip provides primarily Overpower Protection and Core Heat Removal Protection in conjunction with the Overtemperature ΔT trip . The setpoint for this trip is continuously calculated by the RPS using a specific algorithm.

RT is initiated when two out of four loops exceed its setpoint. Figure 7.2-2 sheet 5 shows the logic for this trip function.

(1) Delta T compensation

$$\Delta T \frac{(1+T_{13} s)}{(1+T_{14} s)} \left(\frac{1}{1+T_{15} s} \right)$$

Where: ΔT is the measured RCS ΔT.

s is the Laplace transform operator.

(2) Over power protection setpoint

$$\Delta T_{SP3} = \Delta T_0 \left(K_7 - K_8 \frac{T_6 s}{1+T_6 s} T_{avg} - K_9 \left(T_{avg} - T_{avg0}\right) - f_2(\Delta I) \right)$$

Where: ΔT_{SP3} is the over power protection setpoint.

ΔT_0 is the indicated RCS ΔT at RTP.

s is the Laplace transform operator.

T_{avg} is the measured RCS average temperature.

T_{avg0} is the nominal T_{avg} at rated thermal power.

$f_2(\Delta I)$ is the penalty function of the neutron flux difference between upper and lower part of the power range neutron flux detector. Increase in ΔI beyond a predefined deadband decreases the reactor trip setpoint.

K7, K8 and K9 are coefficient constants.

7.2.1.4.3.3 Low Reactor Coolant Flow

This trip protects the reactor in the event of low reactor coolant flow in one or more loops. RT is initiated when two out of four flow sensors indicate low reactor coolant flow in any loop.

This trip is automatically bypassed when reactor power is below the P-7 permissive setpoint, as indicated by power range neutron flux and turbine inlet pressure. The operating bypass is automatically removed when reactor power is above the P-7 permissive setpoint. Figure 7.2-2 sheet 5 shows the logic for this trip function.

7.2.1.4.3.4 Low Reactor Coolant Pump Speed

This trip protects the reactor core in the event of loss of flow in all loops by tripping the reactor when the speed of two out of four RCPs falls below the setpoint. RCP speed is measured by an electro-magnetic speed detector. This trip is automatically bypassed by permissive P-7. The operating bypass is automatically removed when reactor power is above the P-7 permissive setpoint. Figure 7.2-2 sheet 5 shows the logic for this trip function.

7.2.1.4.3.5 Low Pressurizer Pressure

This trip protects the reactor against low pressure, which could lead to DNB. RT is initiated when two out of four pressurizer pressure channels exceed the low setpoint.

This trip is automatically bypassed when reactor power is below P-7 permissive setpoint (turbine inlet pressure or power range neutron flux). The operating bypass is automatically removed when reactor power is above the P-7 permissive setpoint. Figure 7.2-2 sheet 5 shows the logic for this trip function.

7.2.1.4.4 Primary Over Pressure Protection Trips

7.2.1.4.4.1 High Pressurizer Pressure

This trip protects the RCS against system over pressure. The trip signal is generated when two out of four pressurizer pressure channels exceed the trip setpoint. There are no operating bypasses associated with this trip. Figure 7.2-2 sheet 6 shows the logic for this trip function.

7.2.1.4.4.2 High Pressurizer Water Level

This trip prevents water relief through the pressurizer relief valves for system over pressurization. The trip signal is generated when two out of four pressurizer water level channels exceed the trip setpoint. This trip is automatically bypassed when reactor power is below P-7 permissive. This operating bypass is automatically removed when reactor power is above the P-7 setpoint. Figure 7.2-2 sheet 6 shows the logic for this trip function.

7.2.1.4.5 Loss of Heat Sink Protection

The low SG water level trip protects the reactor from loss of its heat sink in the event of a loss of feedwater to the SGs. The trip signal is generated when two out of four water level sensors, in any SG, monitor water level at or below its trip setpoint. There are no operating bypasses associated with this RT. Figure 7.2-2 sheet 7 shows the logic for this trip function.

7.2.1.4.6 Excessive Cooldown Protection

The high-high SG water level trip protects the reactor from excessive cooldown in the event of excessive feedwater addition to the SGs, and prevents damage to the main turbine by water induction. The trip signal is generated when two out of four water level sensors in any SG exceed the setpoint. This trip is automatically bypassed when reactor power is below the P-7 permissive. This operating bypass is automatically removed when reactor power is above the P-7 setpoint. Figure 7.2-2 sheet 9 shows the logic for this trip function.

7.2.1.4.7 Emergency Core Cooling System Actuation

A trip signal is initiated from each RPS train with actuation of its respective ECCS train, manually or automatically. This trip protects the core against loss-of-coolant or a steam line break. Figure 7.2-2 sheet 11 shows the logic for this trip function.

7.2.1.4.8 Turbine Trip

RT on turbine trip(TT) is an anticipatory trip that is not credited in the safety analysis. Therefore, this is not a safety function but it is designed to be highly reliable. The high reliable design meets the guidance of BTP 7-9(Reference 7.2-12). The RPS and RTB which meet Class 1E criteria with Seismic Category I are applied as the signal processor and final actuation devices for RT on TT. The sensors for RT on TT also meet the requirements of IEEE Std 603-1991. The sensors are located in non-seismic areas (Turbine Building). The installation (including circuit routing) and design of the sensors is such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the reactor protection system and degrade the reactor protection system performance or reliability. Thus the sensors in non-seismic areas are qualified to operate in a seismic event. (i.e., not fail to initiate a trip for conditions which would require a trip.)

RT is initiated by either of following two diverse turbine trip signals:

1. Main Turbine Stop Valve Position

The RT signal is generated within each RPS train when that train receives signals indicating that all four main turbine stop valves are closed. As for all other trips, a RT is generated when two out of four RPS trains have detected this condition.

There are only two limit switches on each main turbine stop valve interfaced to RPS trains A and D, as associated circuits. RPS train A routes the limit switch signals to train B, C, and D. RPS train D retransmits the limit switch signals to train A, B, and C. Interfaces to each RPS train are by isolated digital data links described in ~~Topical Report~~ MUAP-07005 Section 4.3.3 and refer to Figure 7.2-7.

The main turbine stop valve limit switch inputs can be individually bypassed in each RPS train to permit channel testing. Since there are only two limit switch channels, the duration of this maintenance bypass is limited by the technical specifications. This maintenance bypass condition is displayed in the MCR.

This trip is automatically bypassed by permissive P-7 for power level lower than the P-7 setpoint. The operating bypass is automatically removed above P-7 power level. Figure 7.2-2 sheet 13 shows the logic for this trip function.

2. Turbine Emergency Trip Oil Pressure

The turbine emergency trip oil pressure trip signal is generated when two out of four oil pressure channels exceed the trip setpoint. Four oil pressure signals are independently interfaced to each train of the RPS as associated circuits.

This trip is automatically bypassed by permissive P-7 for power level lower than the P-7 setpoint. The operating bypass is automatically removed above the P-7 power level. Figure 7.2-2 sheet 13 shows the logic for this trip function.

7.2.1.5 Manual Control and Actuated Devices

In addition to automatic trip, operators can trip the RTBs using conventional, fixed position, hardwired switches on the OC. There is one switch for each RT actuation train. Actions by under-voltage trip and shunt trip attachments to trip reactor have been discussed in Subsection 7.2.1.2. There are no operating bypasses associated with the manual RT. Maintenance bypasses that allow manual RT testing are described in

MUAP-07004 (Reference 7.2-3) Section 4.4.1. Figure 7.2-2 sheet 4 shows the logic for this trip function.

7.2.1.6 Bypasses

Portions of the safety system can be placed in a bypass mode to allow testing and maintenance while the plant is on-line. Such bypasses are known as maintenance bypasses. Maintenance bypasses are discussed in MUAP-07004.

In addition to maintenance bypasses, automatic and manual operating bypasses are provided to bypass certain protective actions that would otherwise prevent modes of operation such as startup. Automatic and manual operating bypasses are described in Subsections below. Maintenance and operating bypasses may be initiated from safety VDUs. To initiate a maintenance or operating bypass from an Operational VDU, the Bypass Permissive for the train must be enabled.

7.2.1.6.1 Automatic Operating Bypasses

Some operating bypasses are automatically initiated separately within each PSMS train when the plant process permissive condition is sensed by the PSMS input channel(s). Automatically initiated operating bypasses for the RPS are as follows:

- High source range neutron flux trip is bypassed automatically by power range neutron flux (P-10).
- Low reactor coolant flow 1-out-of-4 trip is bypassed automatically during low power conditions (P-7).
- Low RCP speed trip is bypassed automatically by during low power conditions (P-7).
- Low pressurizer pressure trip is bypassed automatically by during low power conditions (P-7).
- High pressurizer water level trip is bypassed automatically by during low power conditions (P-7).
- High-high SG water level trip is bypassed automatically by during low power conditions (P-7).
- Reactor trip on turbine trip is bypassed automatically by during low power conditions (P-7).

All automatically initiated operating bypasses are automatically removed when the plant moves to an operating condition where the protective action would be required if an accident occurred. Refer to Table 7.2-4.

7.2.1.6.2 Manual Operating Bypasses

Some operating bypasses must be manually initiated. These operating bypasses can be manually initiated separately within each PSMS division when the plant process permissive condition is sensed by the PSMS input channel(s). Manually initiated operating bypasses for the RPS are as follows:

- High source range neutron flux trip is bypassed manually with high intermediate range neutron flux (P-6)
- High intermediate range neutron flux trip is bypassed manually with high power range neutron flux (P-10)
- High power range neutron flux (low setpoint) trip is bypassed manually with high power range neutron flux (P-10)

All operating bypasses, either manually or automatically initiated, are automatically removed when the plant moves to an operating regime where the protective action would be required if an accident occurred. Status indication is provided in the MCR for all operating bypasses.

7.2.1.7 Interlocks

Interlocks ensure that operator actions cannot defeat an automatic safety function during any plant condition where that safety function may be required. These interlocks include permissives for manually initiated operating bypasses and interlocks to ensure manually initiated operating bypasses are automatically removed when plant conditions would require the trip functions. Interlocks are also provided to ensure that manually initiated maintenance bypasses can only defeat a single train or channel of the RPS but not multiple channels or trains that would impair the system's ability to function and meet the single failure criteria.

In addition, when safety functions are automatically initiated interlocks, such as reset functions of bistables within the RPS ensure completion of protective actions and ensure that opposing manual actions cannot be taken until acceptable plant conditions are achieved. For example, for most RT functions, when a trip parameter reaches the trip setpoint the partial trip signal will not automatically reset until the plant conditions return to pre-trip conditions. For other trip functions, the partial trip signal must also be manually reset; manual reset cannot occur until after the plant conditions return to pre-trip conditions. Manual reset may be initiated from safety VDUs. To initiate a manual reset from an Operational VDU, the Bypass Permissive for the train must be enabled.

Manual actions that oppose the protective action cannot be taken until three out of four partial trip signals are reset.

7.2.1.8 Redundancy

Redundancy within the RPS is consistent with conformance to the single failure criterion, the unavailability target value and the total safety goal of the plant.

7.2.2.2 Quality of Components and Modules

All safety functions of the RPS are implemented using Class 1E components. Non-safety functions are isolated from the RPS with the exception of the turbine trip inputs, which are treated as associated circuits as discussed in Subsection 7.2.1.4.8.

7.2.2.3 Independence

The independence and separation within the RPS are as described in Subsections 7.1.3.4 and 7.1.3.5, with the exception of the RTBs. Manual RT switches are hardwired from the MCR fire area to the RTBs without isolation. This isolation is not necessary since the reactor will be manually tripped if the MCR must be evacuated due to a fire. Separation and independence of the RTBs are shown in Figure 7.2-5.

7.2.2.4 Defense-in-Depth and Diversity

The diversity features within the RPS are described in Subsection 7.2.1.9. The defense in depth features within the RPS are described in Subsection 7.1.3.1.

7.2.2.5 System Testing and Inoperable Surveillance

Refer to Subsection 7.1.3.14 for details.

7.2.2.6 Use of Digital Systems

All RPS functions rely on digital systems, with the exception of manual RT from the MCR, refer to Subsections 7.1.3.8 and 7.1.3.17. This function uses conventional ~~push buttons~~ switches, which are hardwired directly to electro-mechanical RTBs.

7.2.2.7 Setpoint Determination

The setpoint determination method for the US-APWR is based on the following regulatory guidance, and industry standards:

ANSI/ISA-67.04.01-2000, "Setpoint for Nuclear Safety-Related Instrumentation," February 2000 (Reference 7.2-4).

ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," January, 2000 (Reference 7.2-5).

RG 1.105, Revision 3, "Setpoint for Safety-Related Instrumentation," 1999 (Reference 7.2-6).

Instrument channel statistical accuracy (CSA) and instrument channel response time are integral parts of safety functions setpoint evaluation. The total response time of the instrument channel for the safety function is used in the plant safety analysis. The total response time for safety system actuation signals, and the response time for the actuated devices (which are required to be started to mitigate an AOO or PA), are the basis for calculating acceptable limits for degraded values of the monitored process

7.2.3.1 FMEA Method and Results

The methodology for the FMEA is provided in MUAP-07004 Section 6.5.1. The FMEA follows the guidance of IEEE Std 352-1987 which is referred from IEEE Std 603-1991 and IEEE Std 7-4.3.2-2003, and IEEE Std 379-2000 (Reference 7.2-8) which is endorsed by RG 1.53 (Reference 7.2-9).

Safety functions are designed with multiple divisions. Each safety division is independent from the other safety divisions and from the non-safety division. Independence ensures that ~~credible~~ single failures cannot propagate between divisions within the safety system or between non-safety and safety divisions. Therefore, ~~credible~~ single failures cannot prevent proper protective action at the system level. The ~~credible~~ single failures considered in the non-safety division and safety divisions are described in the FMEA for each system.

The FMEA for reactor trip in PSMS is described in Table 7.2-8 and Figure 7.2-8. The FMEA demonstrates that:

- All ~~credible~~ PSMS failures are detectable (through self-diagnosis or manual surveillance tests).
- No ~~credible~~ single failure will prevent PSMS actuation.
- No ~~credible~~ single failure will result in spurious PSMS actuation, which results in a RT.
- The PSMS will fail to the safe state for all credible failures. The safe state for the RPS is trip. The safe state for the ESFAS/SLS is as-is.

The FMEA is conducted on the basis that one safety channel is continuously bypassed.

7.2.3.2 Safety Analysis

The RT system design requirements such as response time and setpoint determination, are considered and reflected in the safety analysis provided in Chapter 15. The response time, instrument accuracy, and setpoint as shown in Table 7.2-3, meet the safety analysis assumptions.

The Chapter 15 analysis addresses AOOs including spurious control rod withdrawals, plant load rejection, and turbine trip.

Control functions to mitigate the consequences of a plant load rejection and turbine trip are discussed in Section 7.7.

The rod control system interlocks that are provided to prevent abnormal power conditions, which could result from spurious control rod withdrawal, are discussed in Subsection 7.7.1.1.2.

The analysis for additional postulated failures is described as follows:

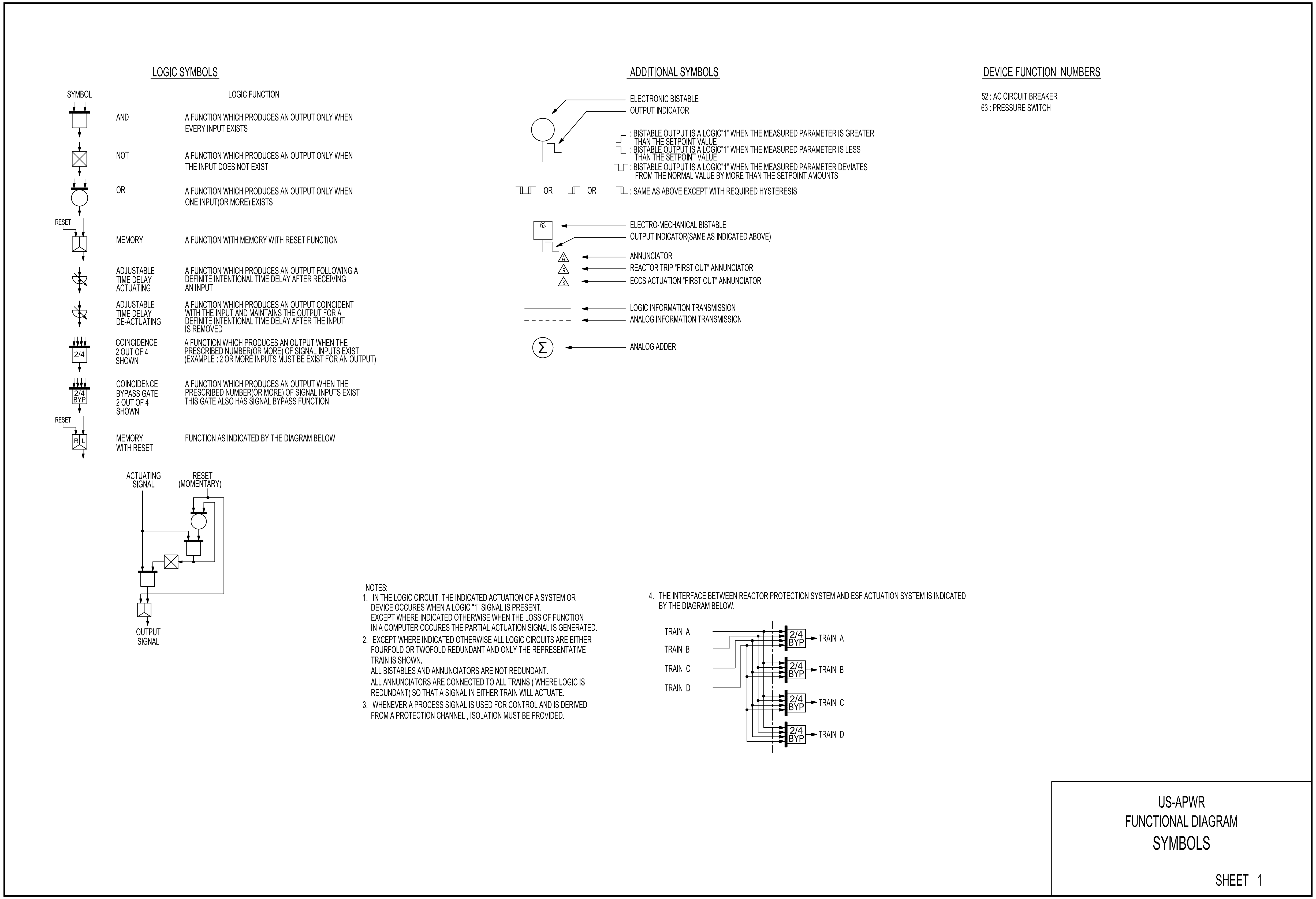


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 1 of 21)

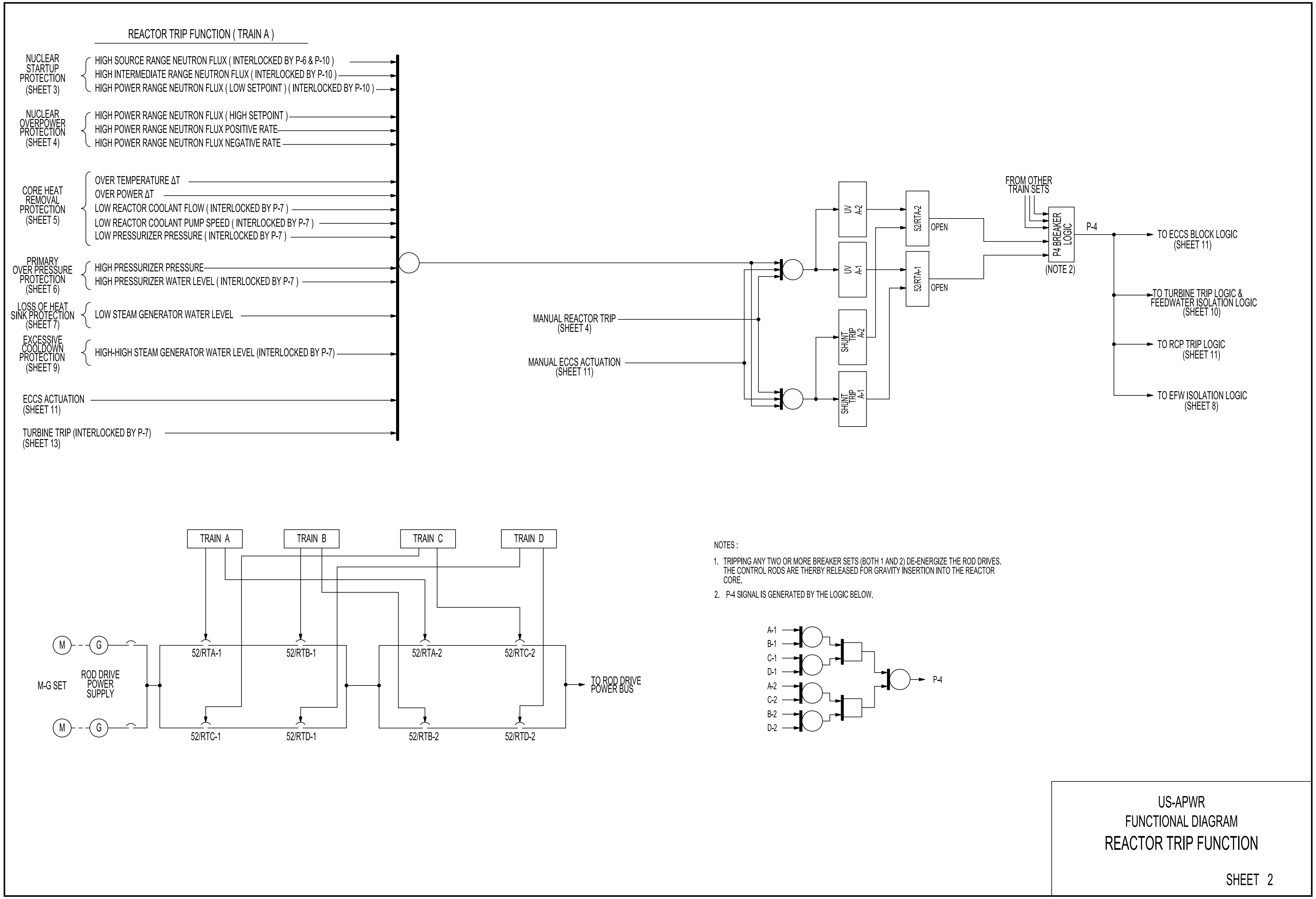


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 2 of 21)

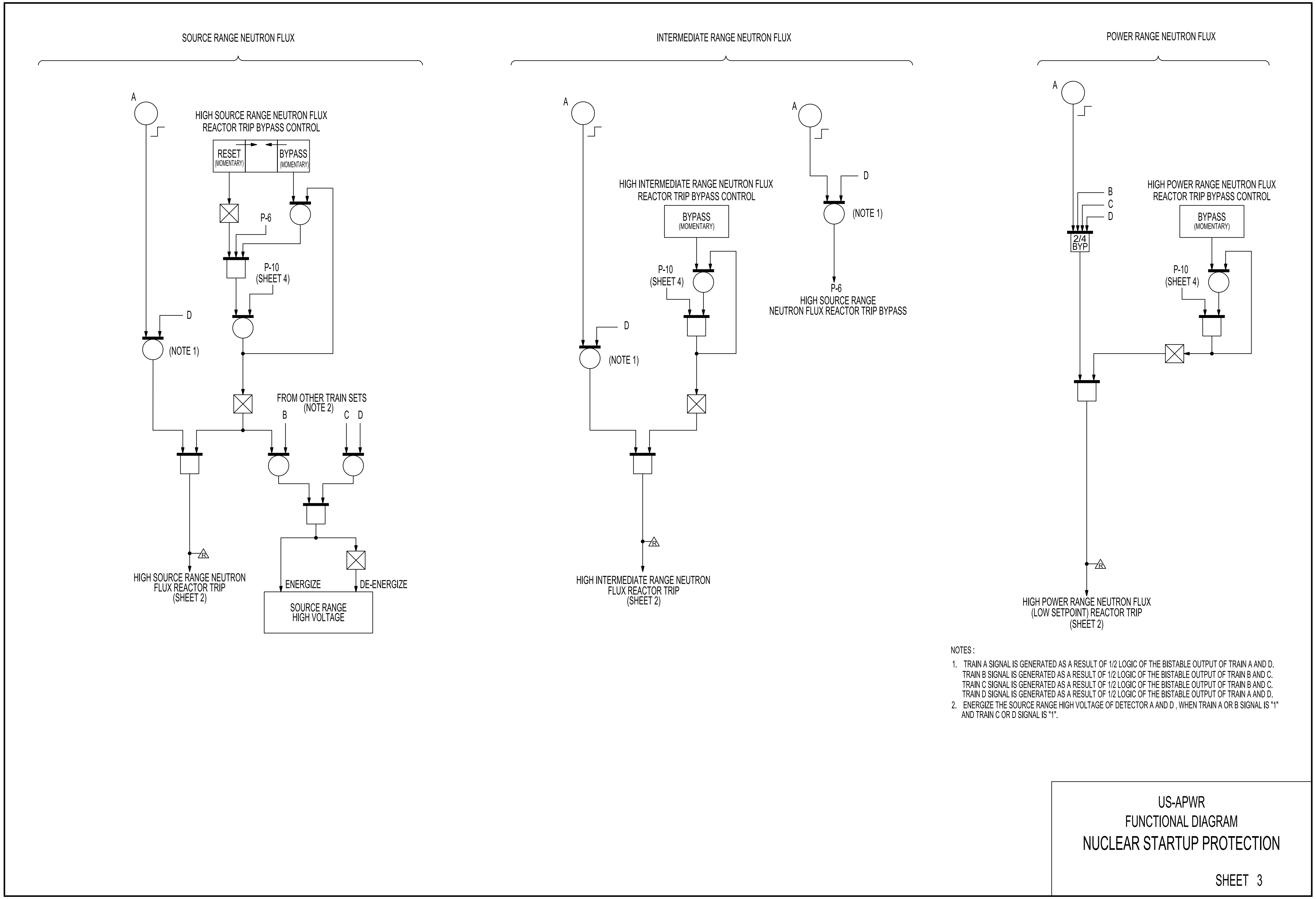


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 3 of 21)

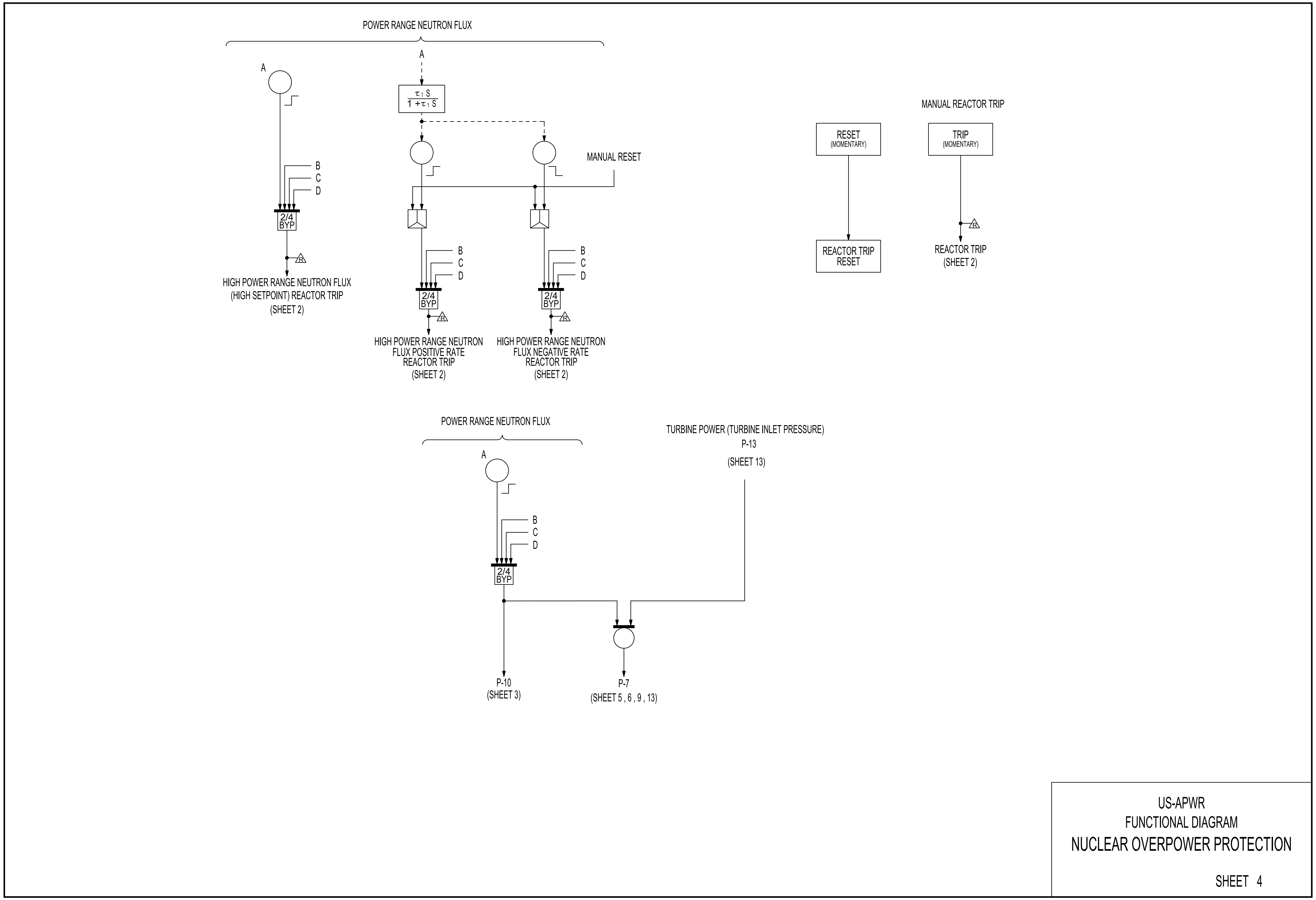


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 4 of 21)

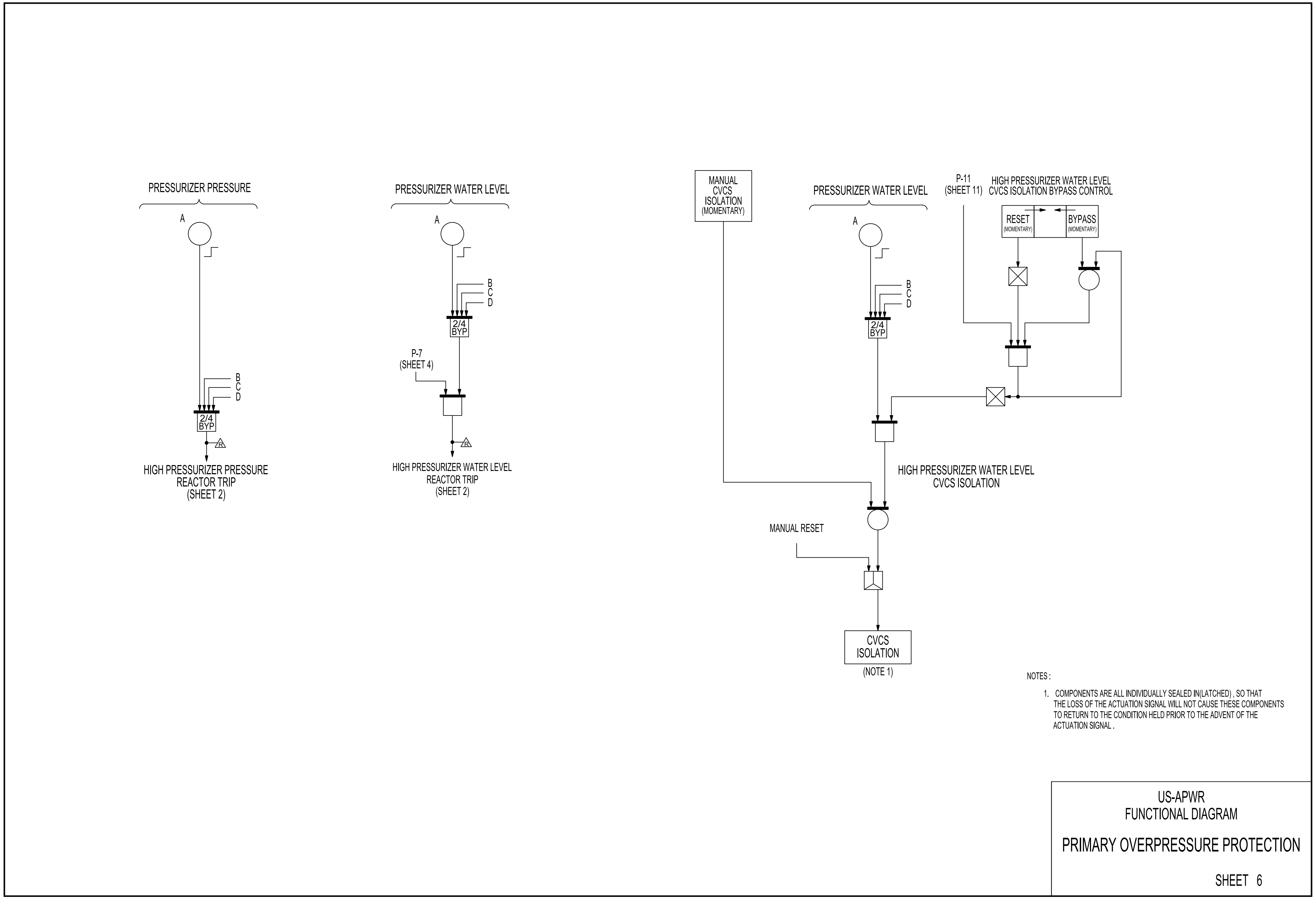


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 6 of 21)

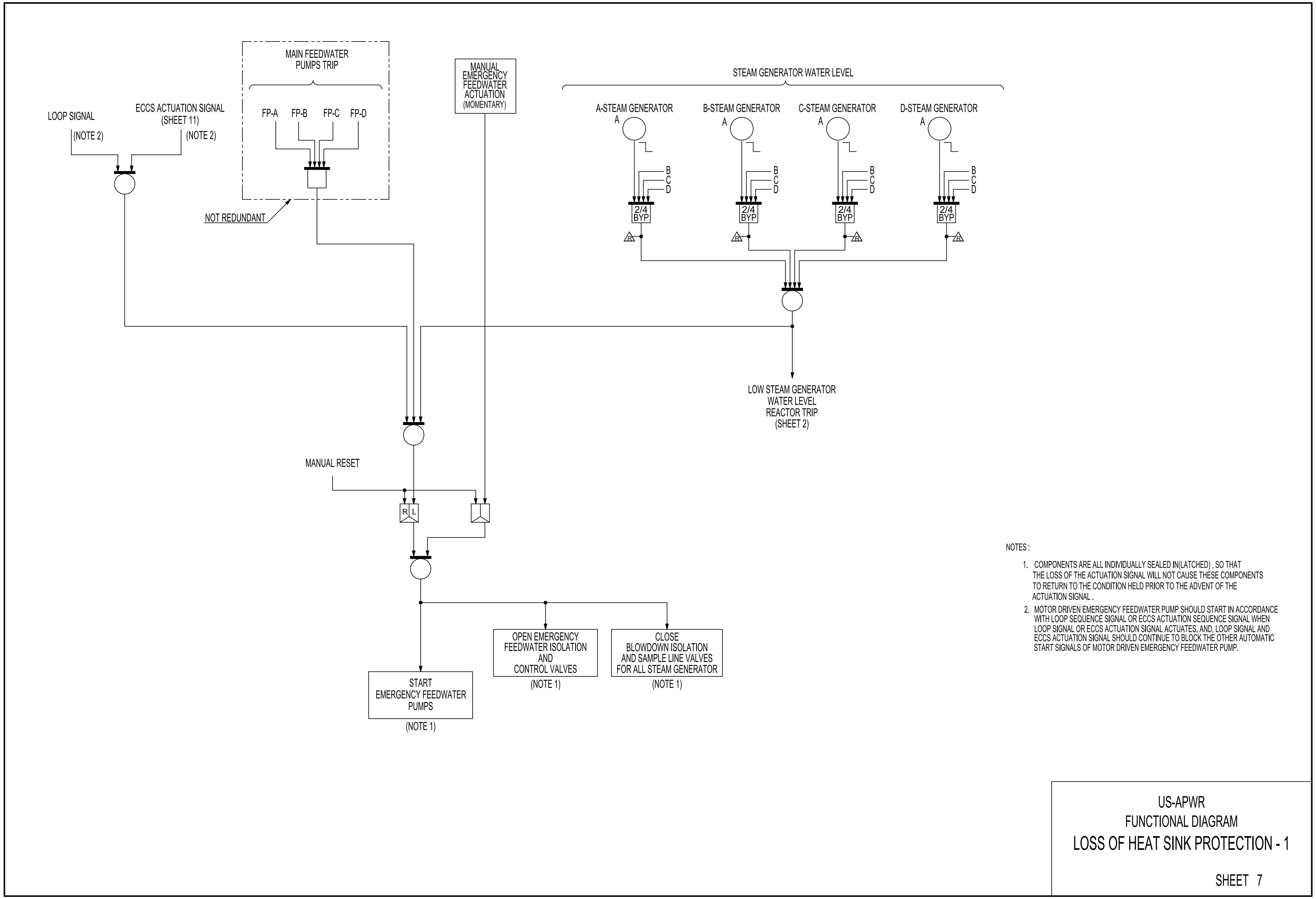


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 7 of 21)

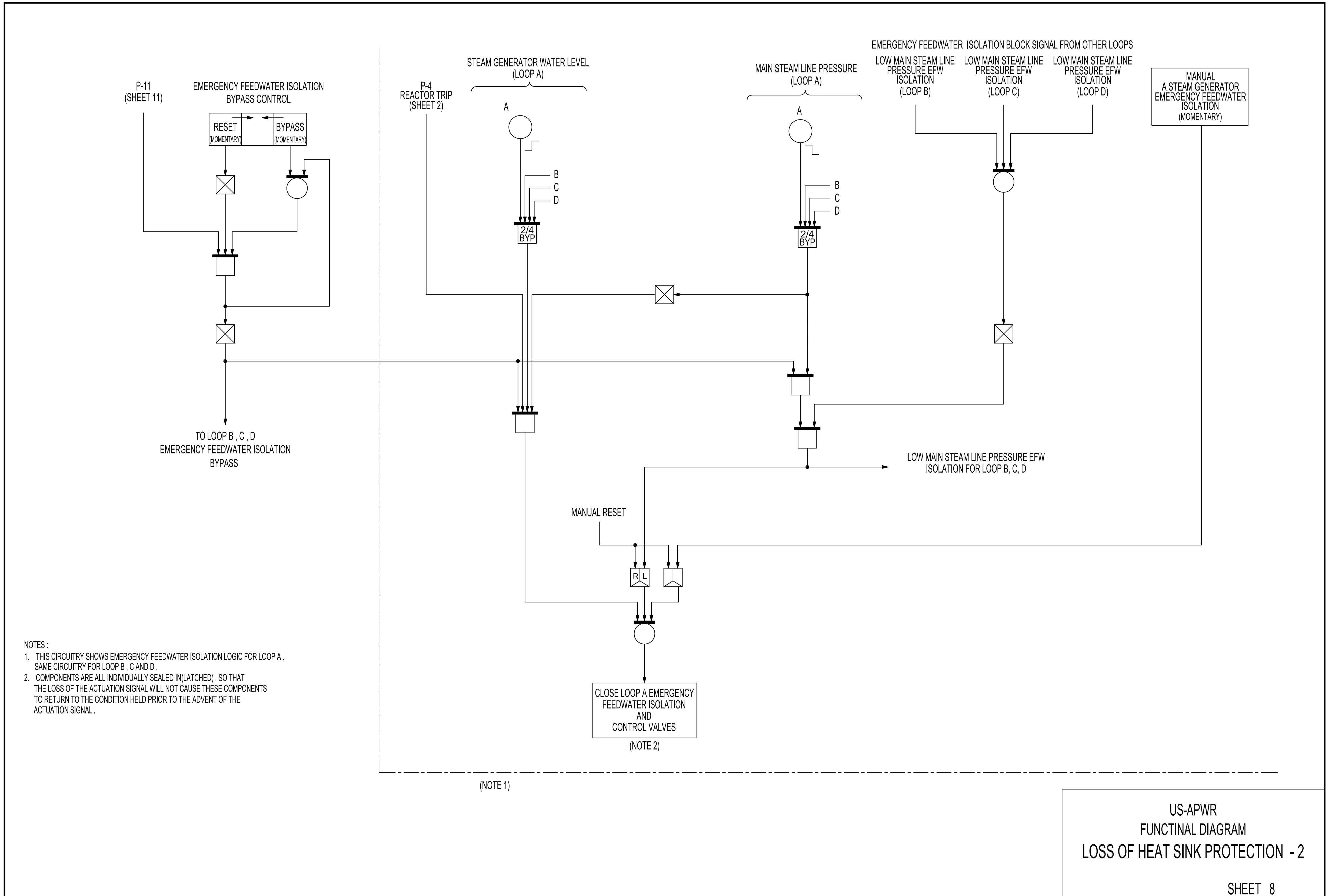


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 8 of 21)

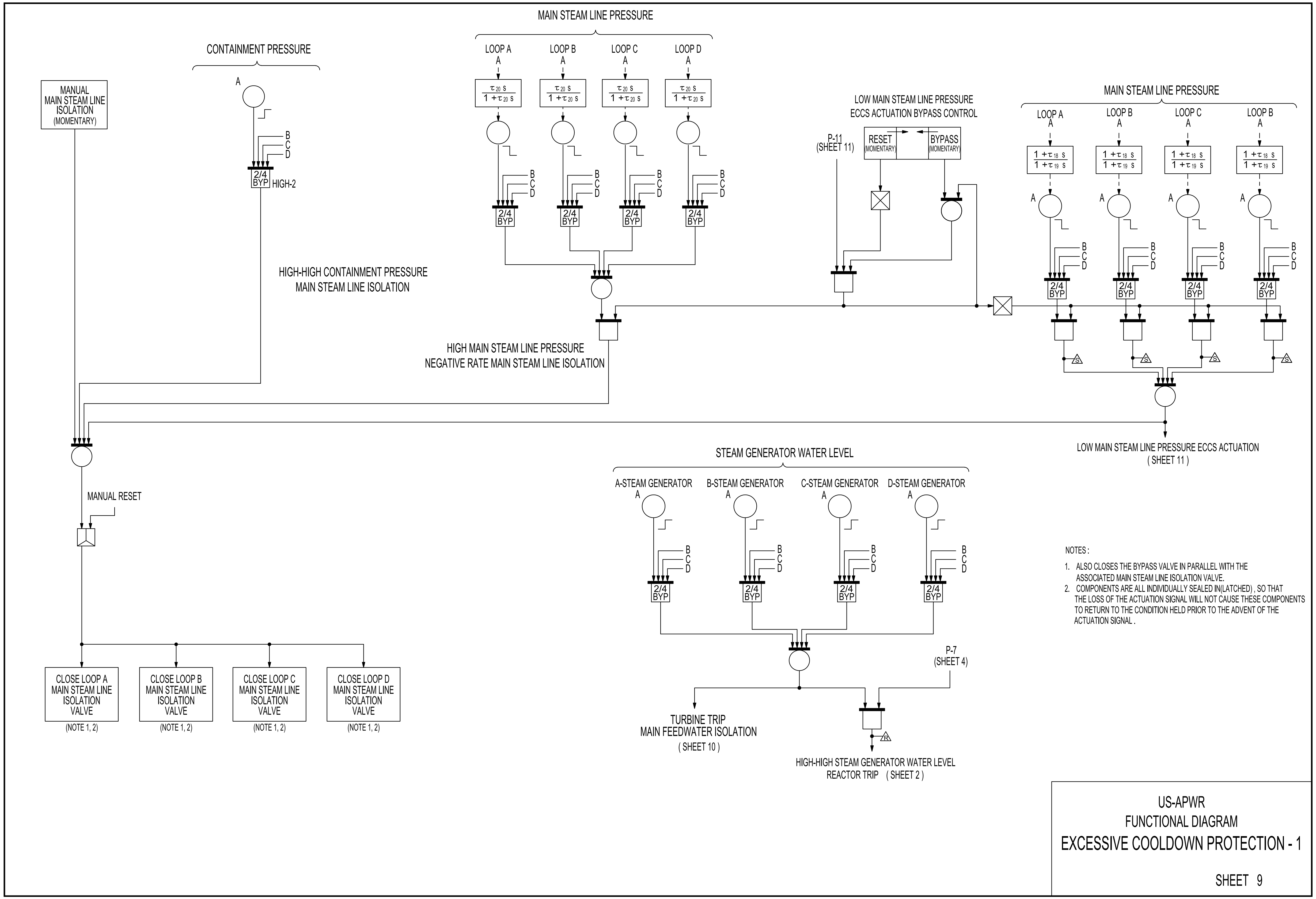


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 9 of 21)

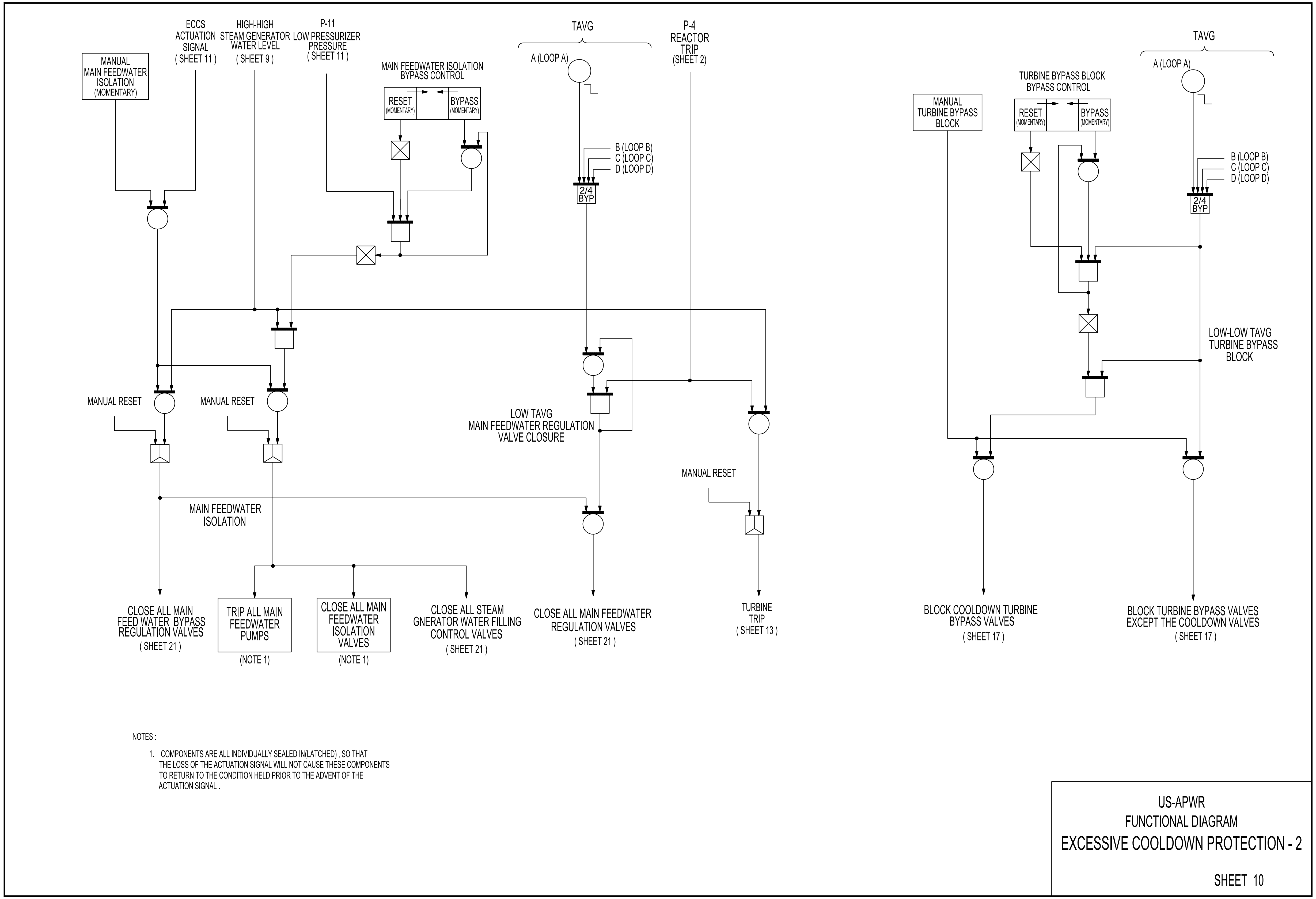


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 10 of 21)

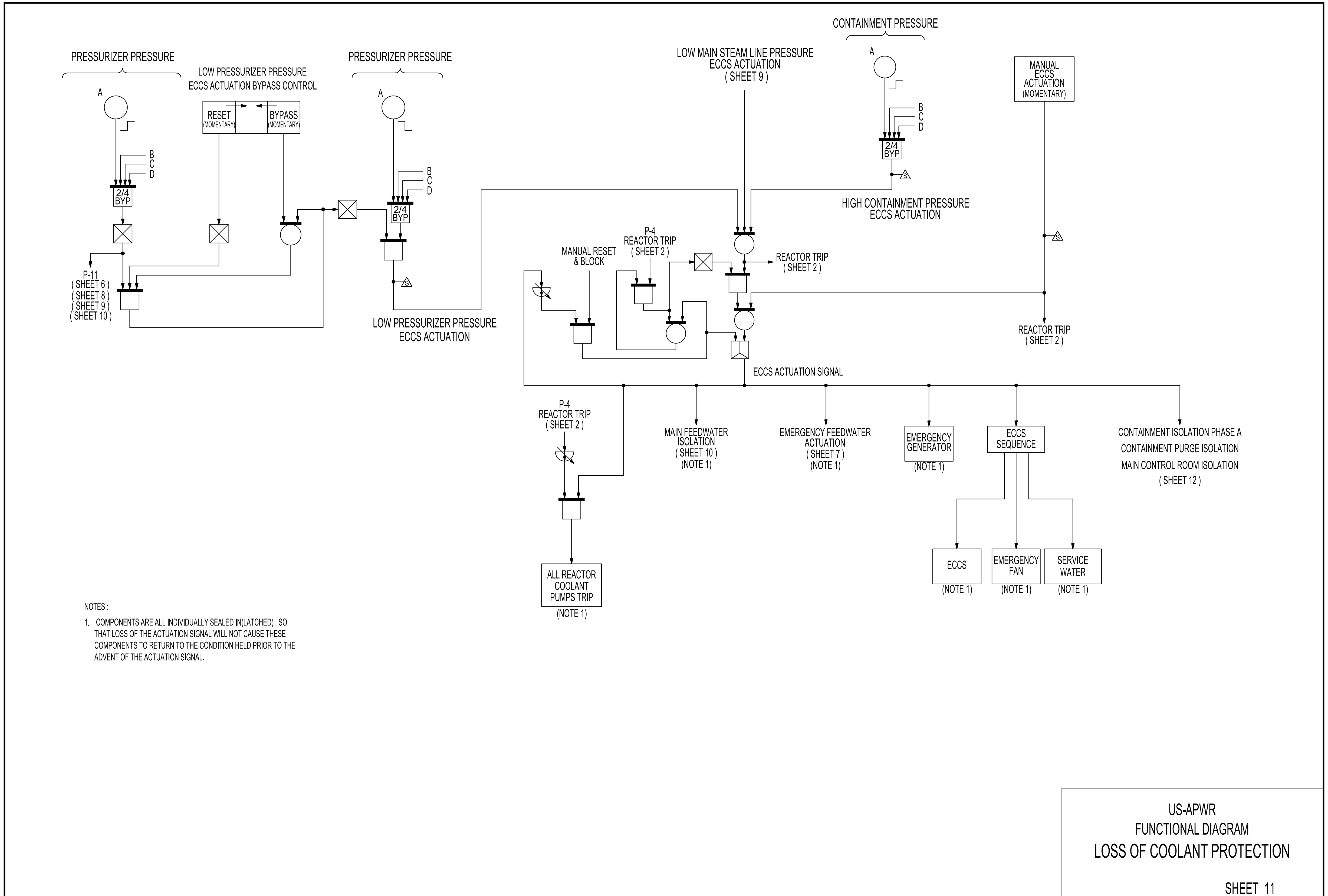


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 11 of 21)

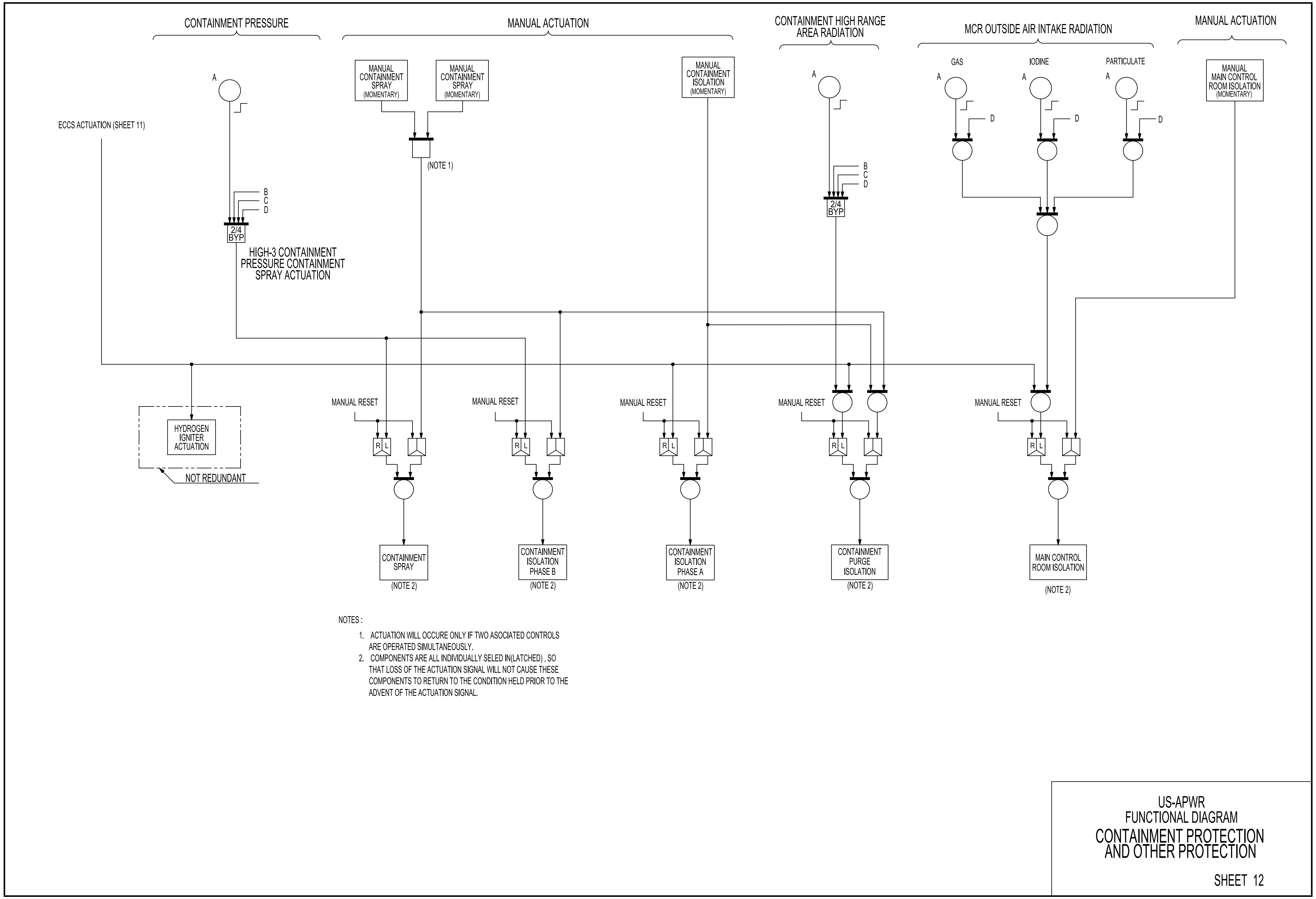


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 12 of 21)

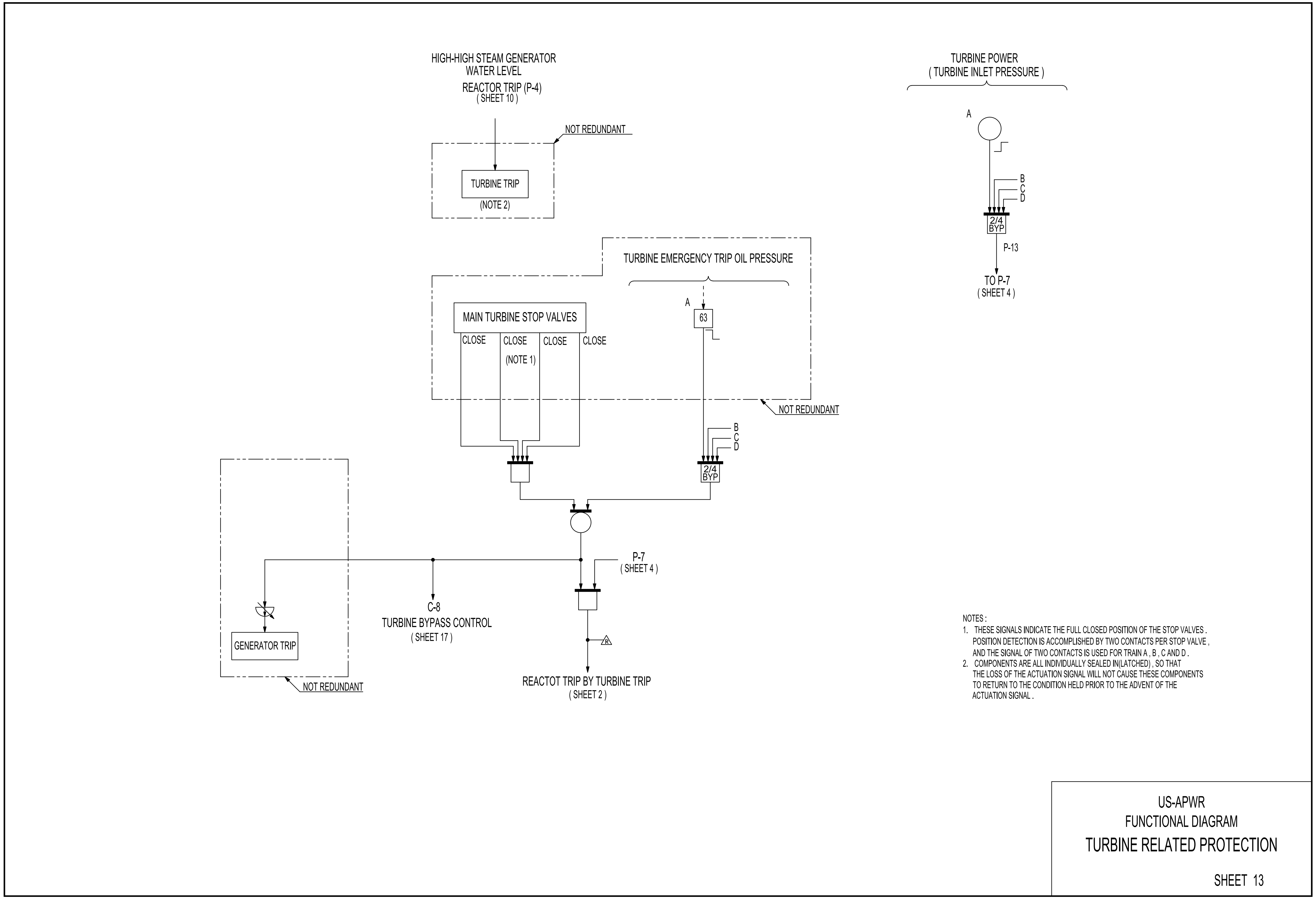


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 13 of 21)

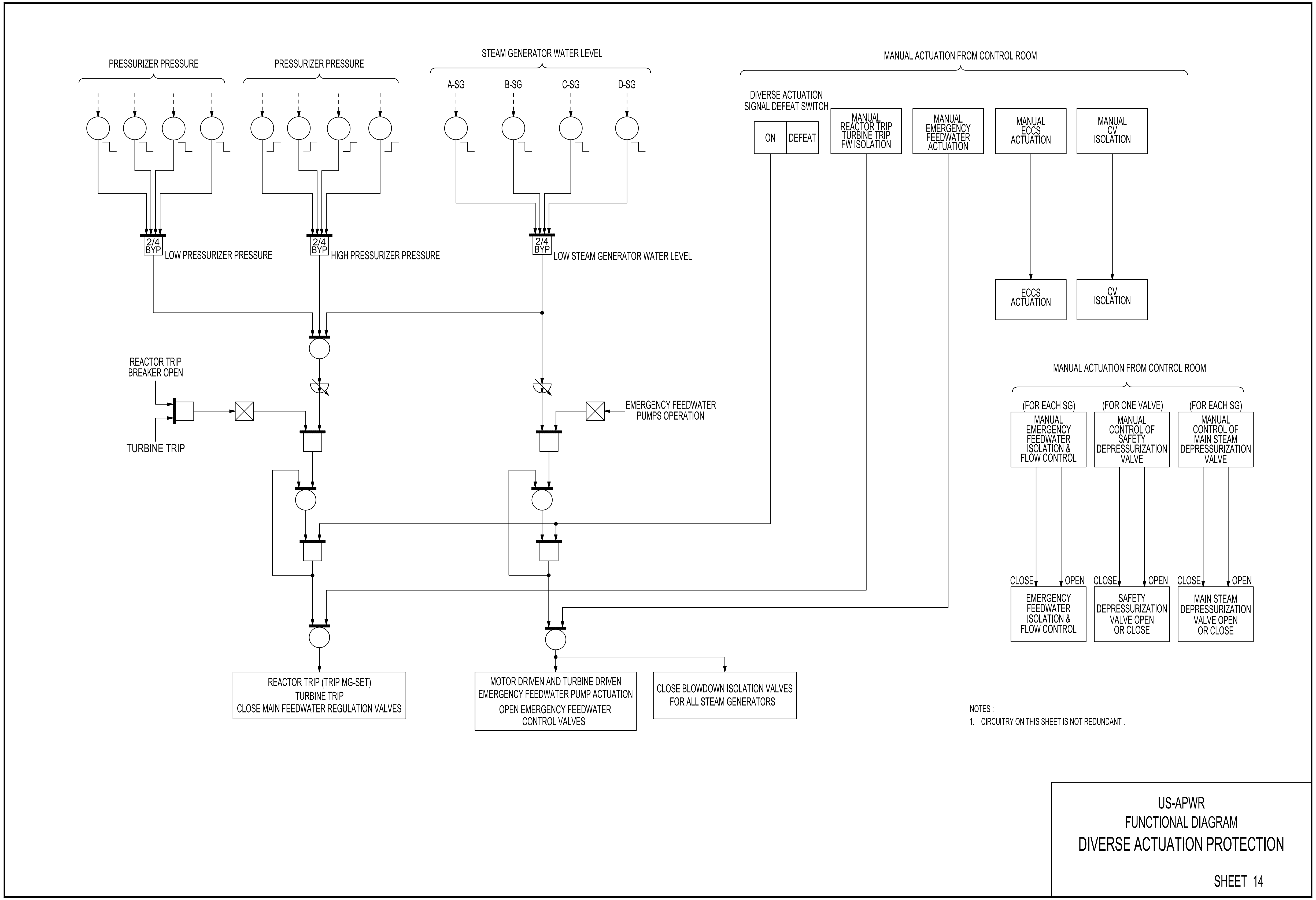


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 14 of 21)

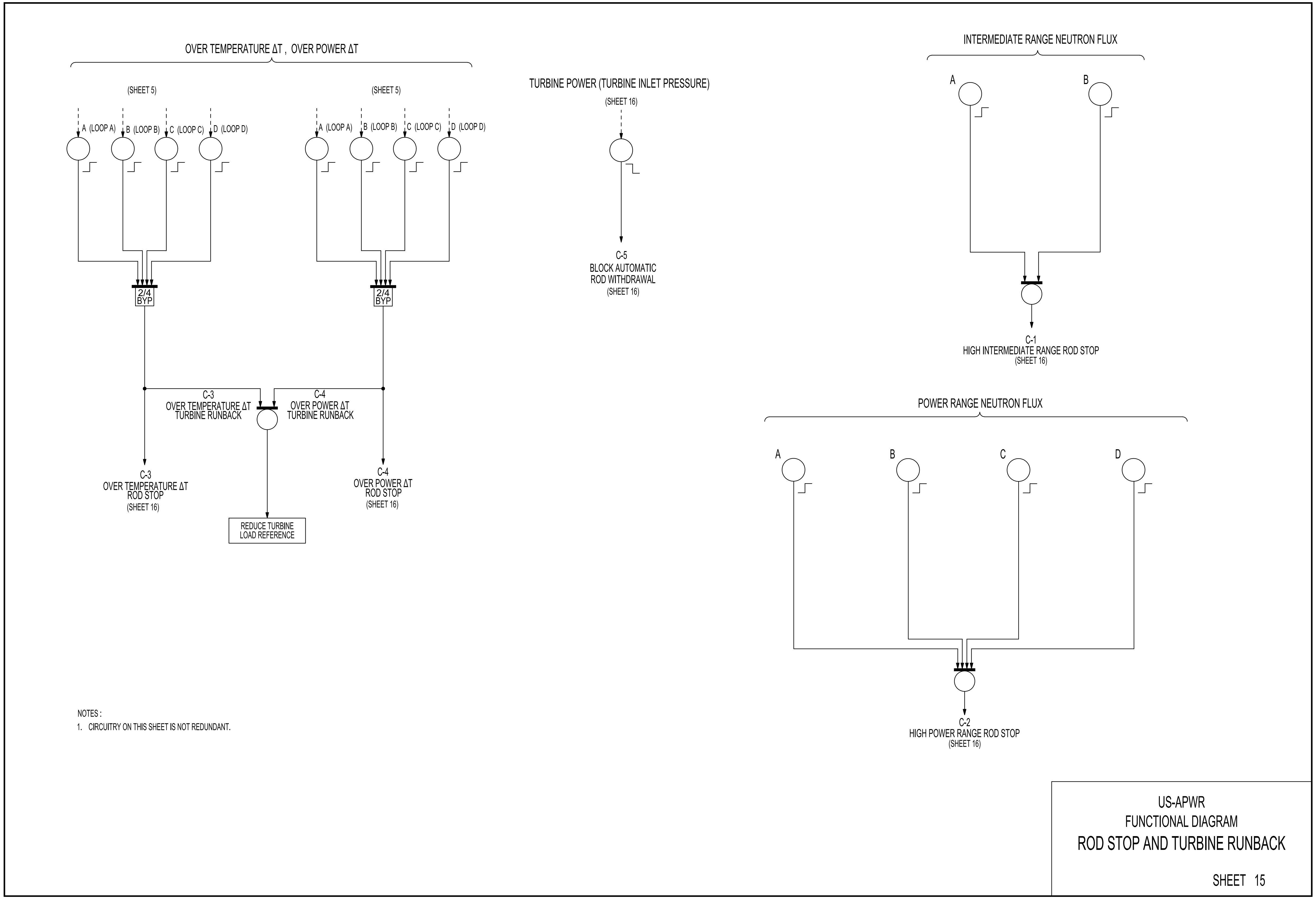
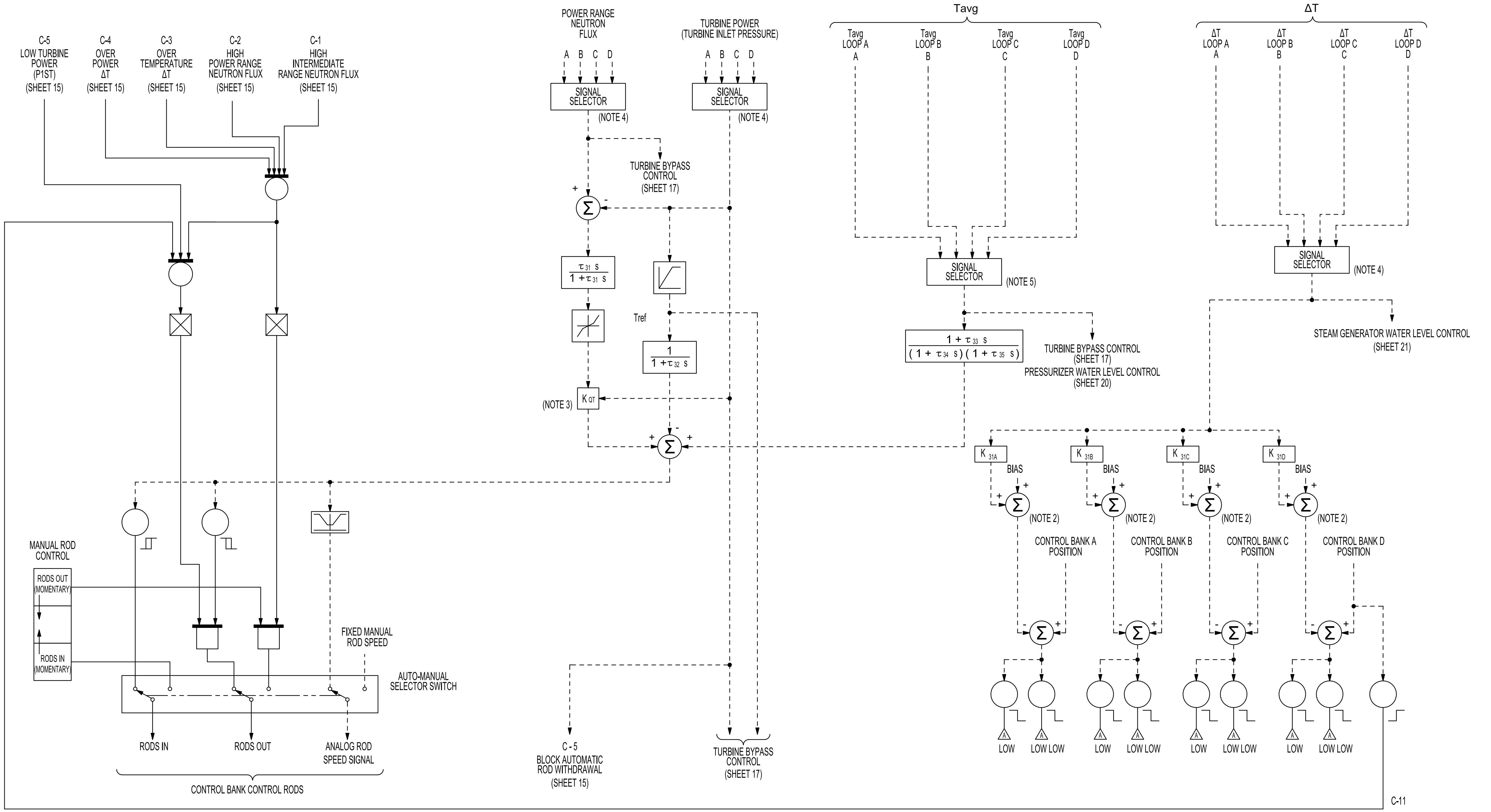


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 15 of 21)



NOTES :
1. CIRCUITRY ON THIS SHEET IS NOT REDUNDANT.
2. THE SUMMER OUTPUTS HAVE FIXED MANUALLY ADJUSTABLE UPPER LIMITS.
3. K_{OT} MAY VARY INVERSELY PROPORTIONAL TO TURBINE LOAD WITH A FIXED LIMIT.
4. THE SECOND HIGHEST SIGNAL IS SELECTED.
5. THE AVERAGE SIGNAL IS SELECTED.

US-APWR
FUNCTIONAL DIAGRAM
ROD CONTROL AND BLOCK
SHEET 16

Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 16 of 21)

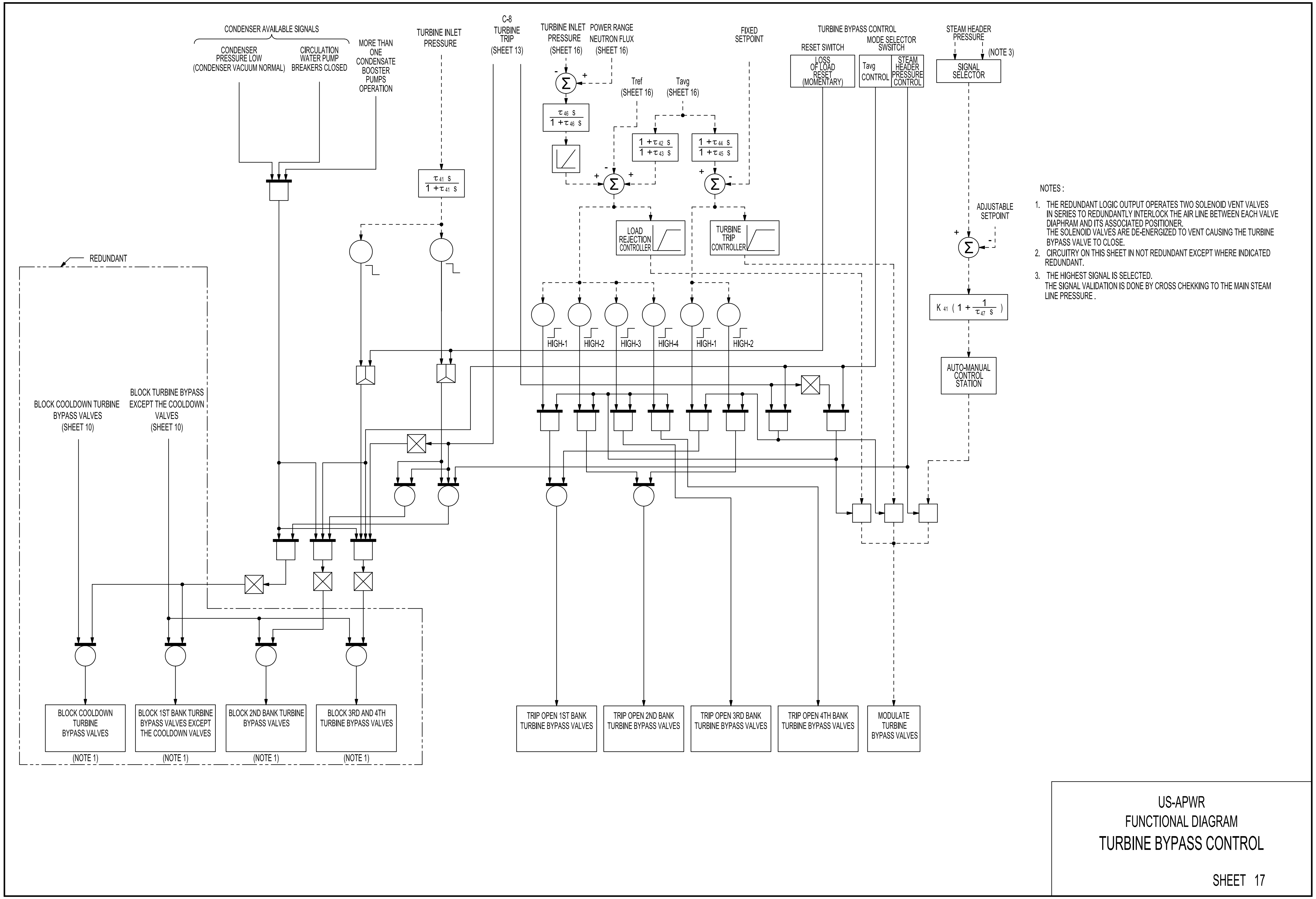


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 17 of 21)

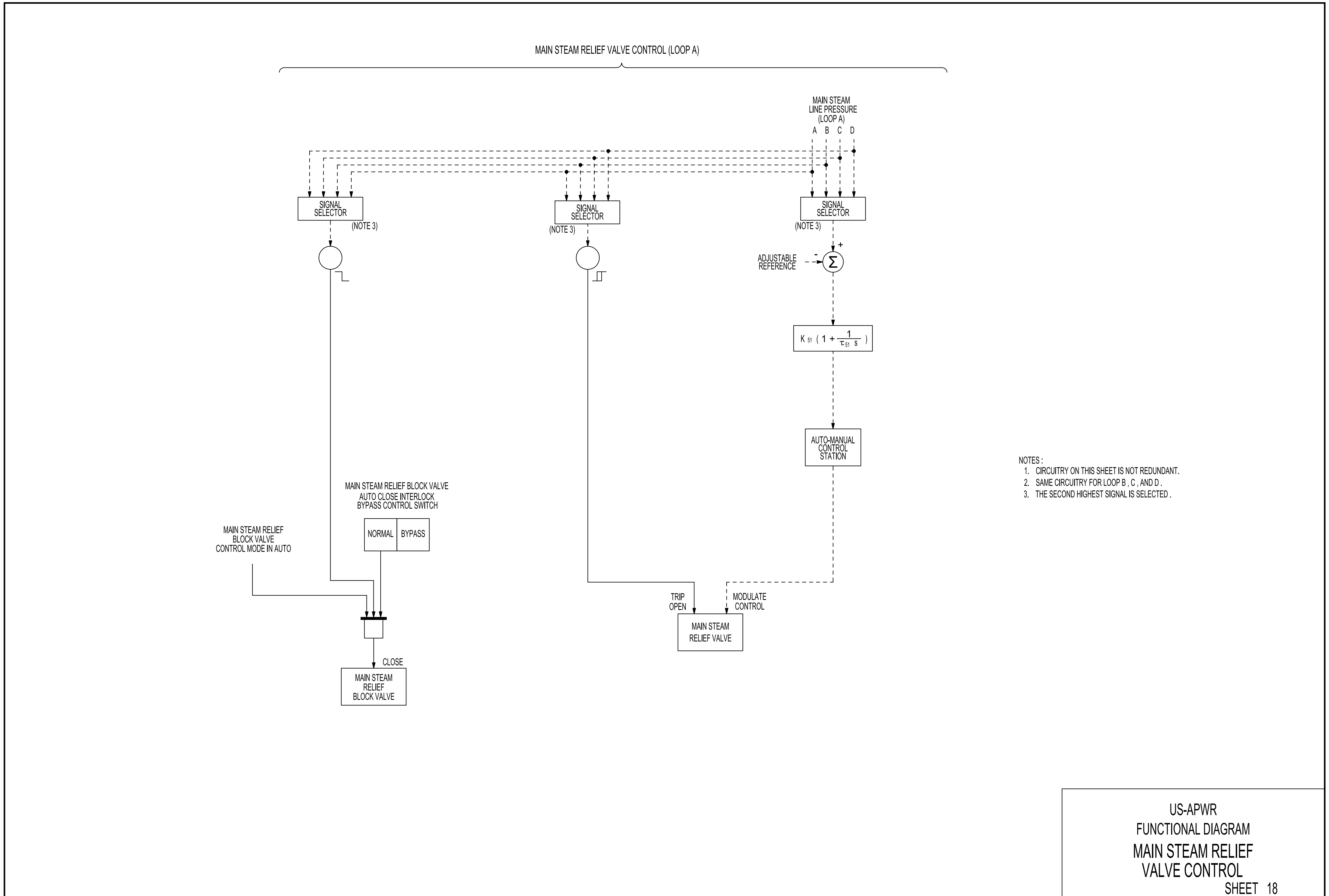


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 18 of 21)

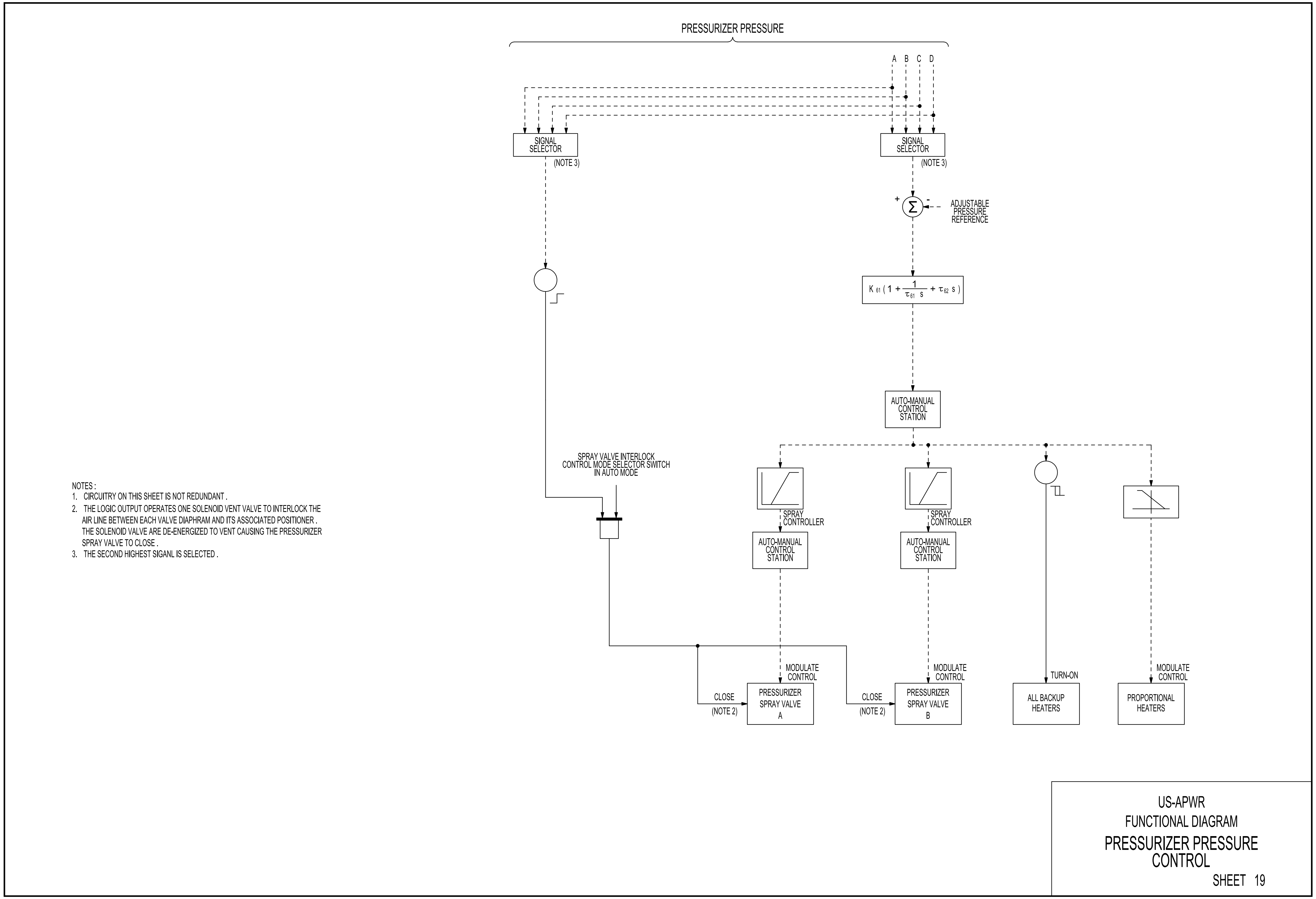
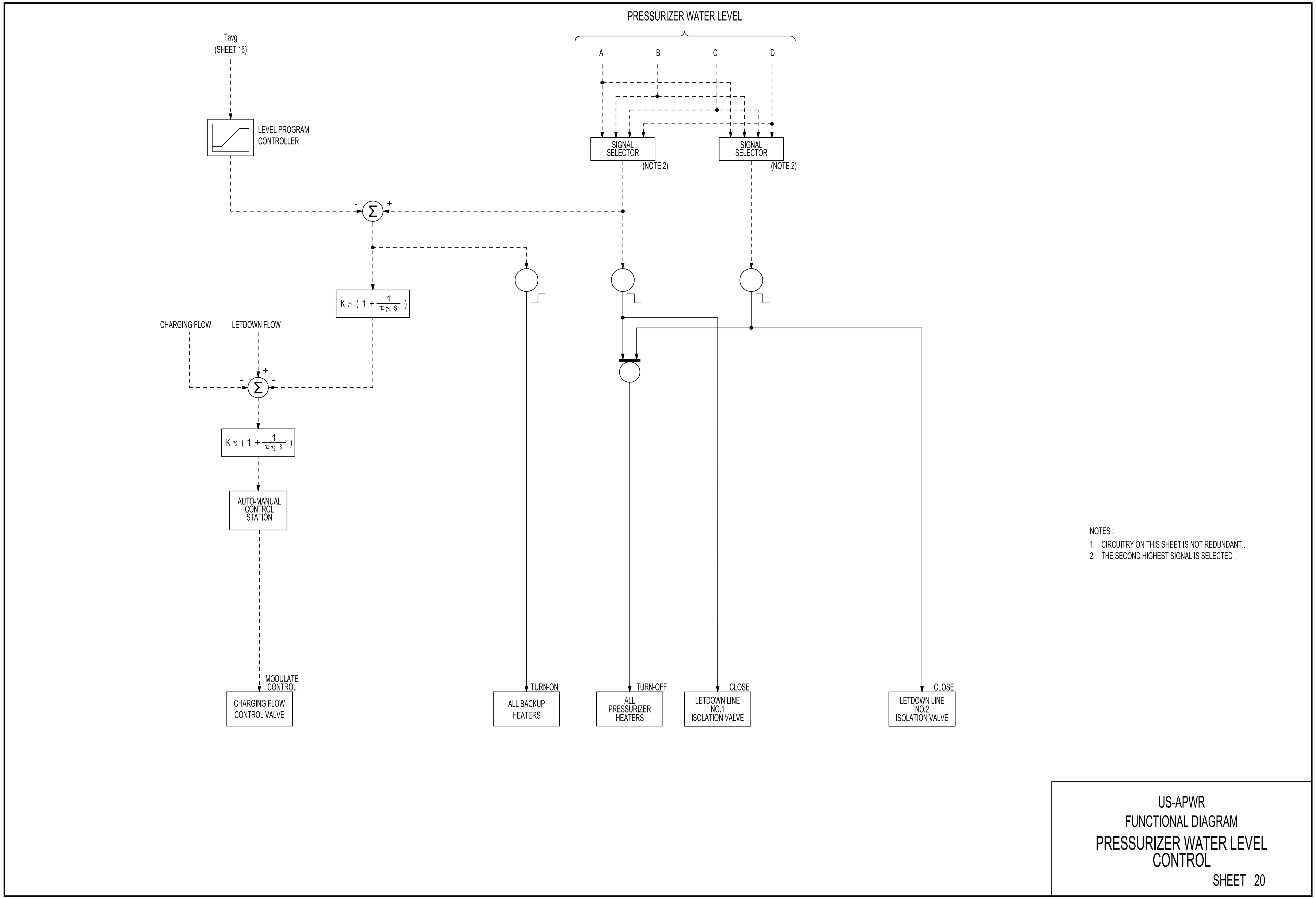


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 19 of 21)



$K_{71}\left(1+\frac{1}{\tau_{71}s}\right)$

$K_{72}\left(1+\frac{1}{\tau_{72}s}\right)$

Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 20 of 21)

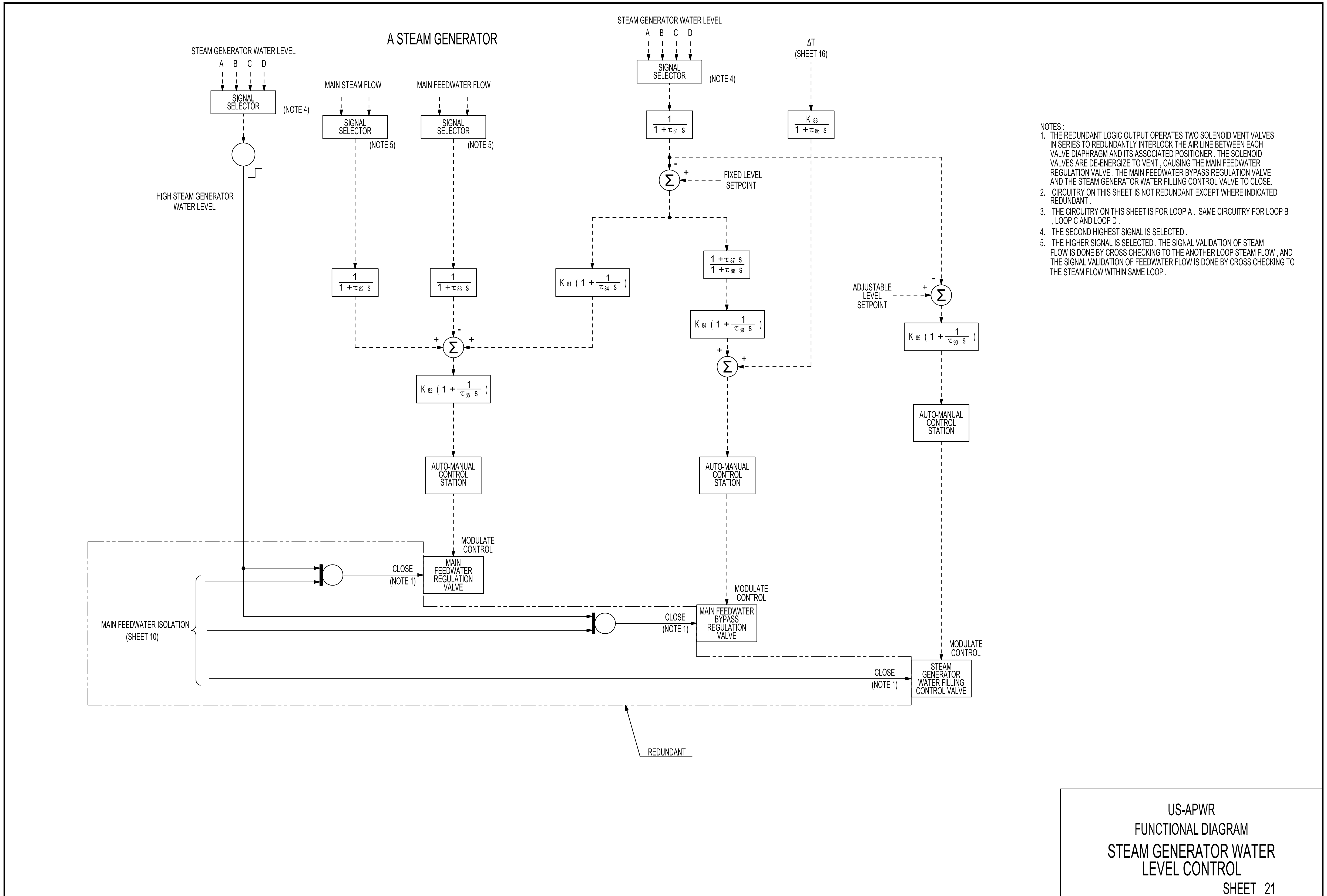
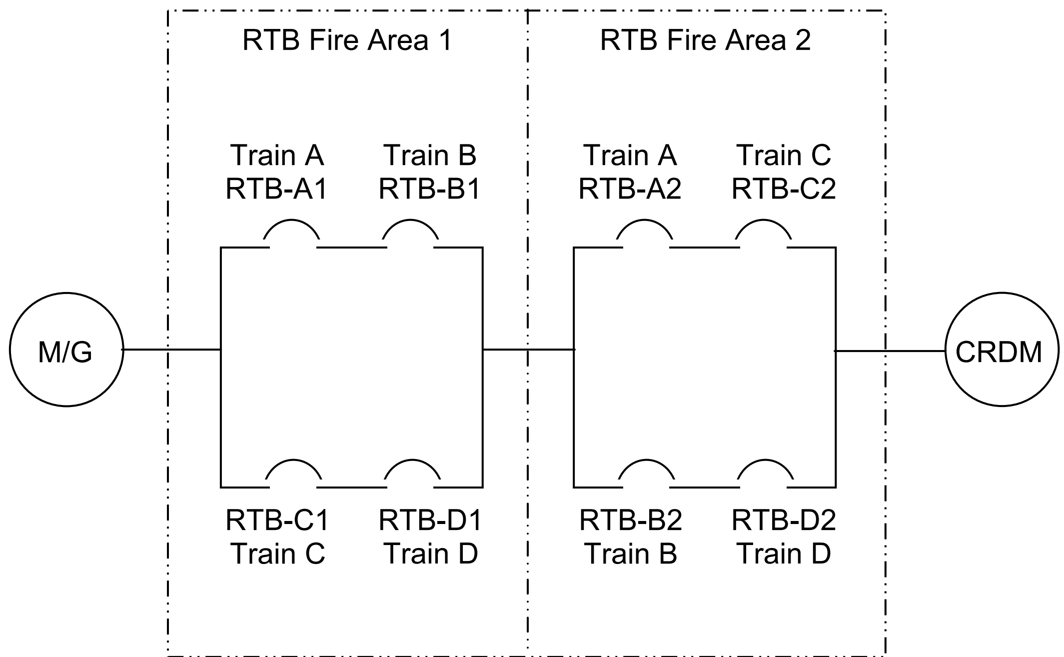


Figure 7.2-2 Functional Logic Diagram for Reactor Protection and Control System (Sheet 21 of 21)



M/G: Motor-Generator Set
CRDM: Control Rod Drive Mechanism

Figure 7.2-4 Configurations of the Reactor Trip Breakers

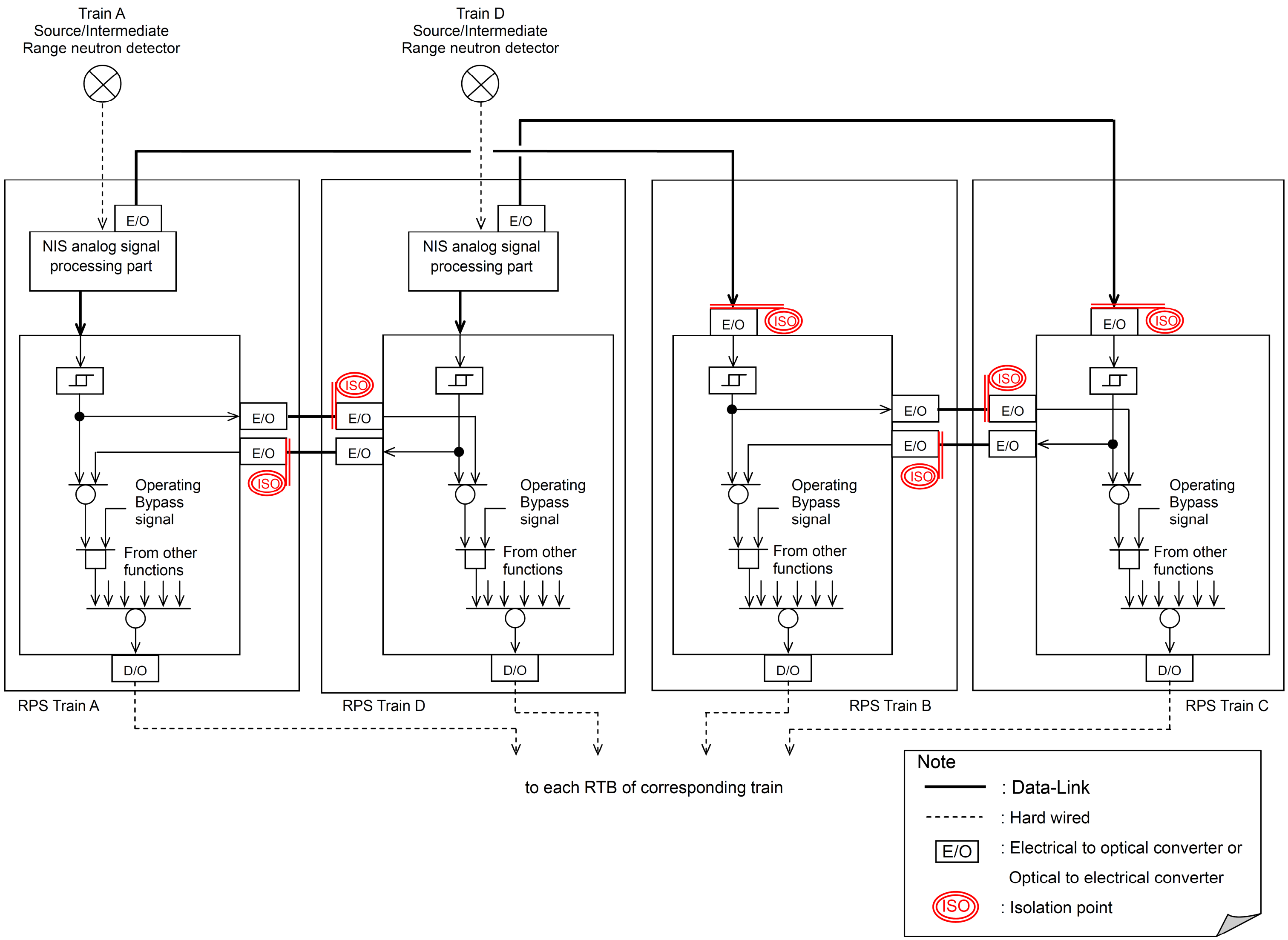


Figure 7.2-6 Signal Flow for High Source and Intermediate Range Neutron Flux Trips

and generates a LOOP signal. Upon detecting a loss of power, the ESFAS starts the Class 1E GTG for its train and disconnects the loads for its train from the electrical bus. Once the Class 1E GTG is capable of accepting loads, the ESFAS sequences the loads for its train back onto the electrical bus in an order appropriate for the current train level ESF actuation signal(s). The ESFAS sequencing logic accommodates ESF actuation signals occurring prior to or during a loading sequence. The ESFAS load sequencing function is independent for each train. The ESFAS also provides automatic load sequencing when an ESFAS is actuated during normal power conditions (i.e., no LOOP). Logic for the ESFAS load sequencing function is described in Subsection 8.3.1.

Safety plant components are manually loaded on the non-safety alternate ac power source from the SLS during station blackout (which includes a loss of the Class 1E GTG Power Source).

7.3.1.2 ESF Component Level Logic

The SLS controls safety-related plant components in all trains based on ESF actuation signals, process instrumentation and component level manual actions from the operational VDUs and safety VDUs.

There are four SLS trains in the US-APWR. The SLS consists of multiple controllers in each train. Plant process systems are assigned to controllers based on consideration of maintenance, potential SLS equipment failures, and optimization of controller performance. In general, complete plant process systems are assigned to a single controller. Multiple process systems are assigned to the same controller or a single process system is assigned to multiple controllers only if the plant effects of controller failure and maintenance are demonstrated to be acceptable, based on ~~the FMEA~~ MUAP-09020 “Function Assignment Analysis for Safety Logic System” (Reference 7.3-11).

Each train of the SLS receives ESF system level actuation demand signals and load sequencing signals from its respective train of the ESFAS. The SLS also receives manual component level control signals from the OC and RSC (safety VDUs and operational VDUs). The SLS also receives process signals from the RPS for interlocks and controls of plant process systems. The system performs the component level control logic for safety actuators (e.g., MOVs, solenoid operated valves, and switchgears).

The SLS controllers for each train are located in separate I&C rooms. The system has conventional I/O portions and I/O portions with priority logic to accommodate signals from the DAS. All SLS I/O will be located within the Class 1E I&C equipment rooms and Class 1E electrical rooms. These rooms are maintained in a mild environment condition by the safety ventilation system at all times.

SLS is a microprocessor based system that achieves high reliability through redundancy within each train and microprocessor self-diagnostics, including data communications. The system also includes features to allow periodic testing of functions that are not automatically tested by the self-diagnostics, such as final actuation of safety components.

Manual periodic tests can be conducted with the plant on-line and without the risk of spurious system level actuation due to single failures during testing.

To enhance reliability, each SLS controller consists of a duplex architecture using redundant CPUs operating in a redundant parallel configuration. In ~~Topical Report~~ MUAP-07005 (Reference 7.3-1), this is referred to as a redundant parallel controller configuration. Each controller of the duplex architecture receives ESF actuation signals and load sequencing signals from the corresponding duplex controller of the ESFAS. The SLS also includes I/O modules mounted in I/O chassis. These I/O chassis can be located within the same cabinet as the controllers, or remotely in separate cabinets that are distributed throughout the plant to reduce the length of cable from the process component or instrument to the I/O chassis. Signals from each SLS controller in the duplex architecture are combined in the output modules using 1-out-of-2 logic for control of plant components to the desired safety state. The SLS I/O modules include contact input conversion devices and power interface devices. The power interface module receives input signals and controls the actuation device (such as motor starters, switchgear, etc.). The actuation devices, in turn, control motive power to the final ESF component. Each train of the SLS thus interfaces the PSMS to each train of the ESF equipment.

Each controller has multiple I/O chassis, each chassis has multiple I/O modules and each I/O module accommodates one or more process interfaces. The plant process interfaces are assigned to I/O modules/chassis with consideration of maintenance and potential SLS equipment failures. Based on the FMEA (refer to Table 7.3-7), acceptable plant level effects for failure or maintenance of any I/O module or any I/O chassis are demonstrated. I/O modules are duplicated within a single SLS train if a single failure of the I/O module will cause a spurious reactor trip.

The primary functions performed by the SLS are described below.

7.3.1.2.1 Control of ESF Components

The ESFAS provides all the system level ESF actuation logic, including the automatic load sequence, for the Class 1E GTG. Whether automatically or manually generated, train level ESF actuation signals are transmitted from each ESFAS train to the corresponding train of the SLS. Within the SLS, the train level ESF actuation signals are then broken down to component actuation signals to actuate each component associated with an ESF. The logic within each train of the SLS accomplishes this function and performs the necessary interlocking to ensure that components are properly aligned for safety.

The SLS also controls ESF components, such as the EFW control valve, based on manual component level controls from operational VDUs and safety VDUs, including all components required for credited manual operator actions, refer to Subsection 7.5.1.5. To ensure spurious command signals from Operational VDUs cannot adversely affect multiple safety divisions, all safety components controlled by the PSMS, regardless of their position under normal operating conditions, are commanded to the correct safety position by automatic safety interlocks or automatic ESFAS actuation signals.

- Low main steam line pressure initiating signal is generated when 2-out-of-4 signals for low pressure in any one of the four loops A, B, C, or D are present and main steam line pressure ECCS actuation bypass is not active. Logic for this actuation circuit is shown on Figure 7.2-2 sheet 9.

The low pressurizer pressure ECCS actuation bypass and low main steam line pressure ECCS actuation bypass can be activated manually only when pressurizer pressure interlock P-11 is present (i.e., when the pressurizer pressure signal is lower than the P-11 setpoint). These manually initiated operating bypasses are automatically removed when the pressurizer pressure signal is higher than the P-11 setpoint.

- High containment pressure initiating signal is generated when 2-out-of-4 signals for high containment pressure are present. There is no operating bypass associated with this ECCS actuation signal. Logic for this actuation circuit is shown on Figure 7.2-2 sheet 11.

An activated ECCS signal is latched separately for each train and cannot be manually overridden for 160 seconds. After ECCS is manually overridden the override is automatically removed when the P-4 RT interlock clears (i.e., RTB re-closed). An ECCS actuation signal cannot be manually reset for 160 seconds after actuation and until the initiating signals have cleared.

An ECCS actuation signal aligns the required ESF systems valves (e.g., containment isolation valves, EFW valves) and starts the ESF system pumps and fans, required to mitigate the specific accident and/or AOO conditions. An ECCS actuation signal results in the following actions:

- Trip RCPs: There are two Class 1E RCP breakers for each RCP. One breaker is located in the Class 1E electrical room and the other is located in the electrical room in the turbine building. All Class 1E RCP breakers are tripped in 15 seconds after both the ECCS actuation signal and the P-4 RT interlock signal are present. The P-4 interlock is generated when breaker open status signals are received from any combination of RTBs that would result in a RT. Logic for this actuation is included on Figure 7.2-2 sheet 11.

- Start emergency generator: Actuation of ECCS signal starts the emergency power source.

- Safety injection pumps

- ~~Containment spray/residual heat removal (CS/RHR) pumps~~

- RT: RT is initiated by the ECCS actuation signal, refer to Section 7.2.

- Main feedwater isolation

- Emergency feedwater actuation

- Containment isolation phase A

isolation bypass control is common to all SG loops A, B, C, and D, whereas there are separate EFW isolation bypass controls for each ESFAS train.

7.3.1.5.11 CVCS Isolation

There are two ESFAS trains for chemical and volume control system (CVCS) Isolation, train A and train D.

The ESF actuation signal for the CVCS isolation function is generated on a condition when any of the following initiating signals are present:

- Manual actuation
- High pressurizer water level: this signal is present when 2-out-of-4 signals are present for high pressurizer water level and high pressurizer water level CVCS isolation bypass control (operating bypass) is not present.

The CVCS isolation bypass control (operating bypass) can only be actuated when P-11 interlock is present. This operating bypass is automatically removed by the P-11 interlock when pressurizer pressure rises above the P-11 setpoint.

The resulting actuation signal is latched. Logic diagram for this function is included on Figure 7.2-2 sheet 6.

~~7.3.1.5.12Block Turbine Bypass and Cooldown Valves~~

~~There are two ESFAS trains for block turbine bypass and cooldown valves, train A and train D.~~

~~The block turbine bypass valve signal is initiated on the following conditions:~~

- ~~Low-low T_{avg} signal: This signal is present when 2-out-of-4 RCS loops indicate low-low T_{avg}.~~
- ~~The manual turbine bypass block switch is selected to close (manual actuation).~~

~~The resulting block turbine bypass valve signal blocks the opening of the turbine bypass valves to the condenser. This block does not affect the turbine bypass valves to the condenser, which are referred to as the “turbine bypass cooldown valves.” The block is automatically removed when the low-low T_{avg} signal goes above its setpoint.~~

~~Logic for this function is included on Figure 7.2-2 sheet 10.~~

~~The block cooldown turbine bypass signal initiates on the same low-low T_{avg} signal and manual actuation, as described above. The resulting block cooldown turbine bypass signal blocks the turbine bypass cooldown valves to the condenser. The block is automatically removed when the low-low T_{avg} signal goes above its setpoint.~~

~~The block cooldown turbine bypass signal may be manually overridden by the turbine bypass block bypass control switch (operating bypass). The override can only be actuated after the signal is actuated from low-low T_{avg}. The override is automatically reset when the low-low T_{avg} signal returns to normal. The override may be manually reset from the turbine bypass block bypass control switch. The block turbine bypass override has no effect on the block turbine bypass valve signal (i.e., it only affects the cooldown signal).~~

~~Logic for this function is included on Figure 7.2-2 sheet 10.~~

~~The turbine bypass valves and the turbine bypass cooldown valves are controlled by two permissive solenoids on each valve, train A and D respectively.~~

7.3.1.6 Bypasses and Overrides

The safety system can be placed in a bypass mode to allow testing and maintenance while the plant is on-line. Such bypasses are known as maintenance bypasses. Maintenance bypasses are discussed in MUAP-07004 (Reference 7.3-2). In addition to maintenance bypasses, automatic and manual operating bypasses are provided to block certain protective actions that would otherwise prevent modes of operations such as startup. Automatic and manual bypasses are described in the following subsections. Maintenance and operating bypasses may be initiated from safety VDUs. To initiate a maintenance or operating bypass from an Operational VDU, the Bypass Permissive for the train must be enabled.

7.3.1.6.1 ESF System Maintenance Bypass

Bypasses are provided in each ESF system train to block the actuation of one or more ESF signals (e.g., ECCS actuation, EFW, main steam isolation, etc.). The purpose of these bypasses is to allow maintenance on an ESF process system, or to accommodate an ESFAS/SLS controller failure. There are alarms for ESF systems out of service conditions that block functionality at the train level.

7.3.1.6.2 Automatic Operating Bypasses

These operating bypasses are automatically initiated separately within each PSMS division when the plant process permissive condition is sensed by the PSMS input channel(s). The following is a list of automatically initiated operating bypasses:

- High main steam line pressure negative rate initiating signal for main steam line isolation is automatically bypassed when the P-11 interlock clears (when pressurizer pressure is above the setpoint). This operating bypass can be manually removed when the P-11 is present (when pressurizer pressure is below the setpoint).
- When the P-4 interlock clears (RTB closed) the low T_{avg} initiating signal for main feedwater isolation (for closing all main feedwater regulation valves) is automatically bypassed. This operating bypass is automatically removed when the P-4 interlock is present (RTB open).

7.3.1.6.3 Manual Operating Bypasses

Some operating bypasses must be manually initiated. These operating bypasses can be manually initiated separately within each PSMS train when the plant process permissive condition is sensed by the PSMS input channel(s). The following is a list of manually initiated operating bypasses:

- Low pressurizer pressure initiating signal for the ECCS actuation function can be manually bypassed only when the P-11 interlock is present (pressurizer pressure is below the setpoint). This operating bypass is automatically removed when the P-11 interlock clears (when pressurizer pressure is above the setpoint).

- Low main steam line pressure initiating signal for the ECCS actuation function and main steam line isolation function can be manually bypassed only when the P-11 interlock is present (pressurizer pressure is below the setpoint). This operating bypass is automatically removed when the P-11 interlock clears. When this operating bypass is active, the high main steam line pressure negative rate trip is enabled.

- MFW isolation function can be bypassed manually only when the P-11 interlock is present. This operating bypass is automatically removed when the P-11 interlock clears.

- The EFW isolation function actuated by low main steam line pressure can be manually bypassed if the P-11 interlock is present. This operating bypass is automatically removed when the P-11 interlock clears.

- The manual bypass for high pressurizer water level initiation signal for CVCS isolation can only be actuated when the P-11 interlock is present. This operating bypass is automatically removed when the P-11 interlock clears.

All operating bypasses, either manually or automatically initiated, are automatically removed when the plant moves to an operating condition for which the protective action would be required if an accident occurred. Status indication is provided in the MCR for all operating bypasses.

7.3.1.6.4 Manual Overrides

Manual overrides must be manually initiated. These manual overrides can be manually initiated separately within each PSMS train when the plant process permissive condition is sensed by the PSMS input channel(s). The following is a list of ~~system~~ train level manually initiated overrides:

- The ECCS actuation can be manually overridden at the ~~system~~ train level when the P-4 interlock is present (RTB open). This manual override is automatically removed when the P-4 interlock clears (RTB closed). In MUAP-07004 Appendix D (e), this override is referred to as a reset.

- The block cooldown turbine bypass valve actuation by low-low T_{avg} may be manually overridden at the ~~system~~ train level. This manual override cannot be initiated until after automatic system level actuation. The manual override may be manually reset by the operator at any time, and is automatically reset when the low-low T_{avg} initiation signal returns to normal. This signal blocks the cooldown turbine bypass valves. In MUAP-07004 Appendix D (b), this override is referred to as an operating bypass.

7.3.1.7 Interlocks

The interlocks for initiating and automatically removing operating bypasses are discussed above. The interlocks for manual overrides are discussed above. The interlocks for resetting system level actuation and channel level actuation are discussed in Subsection 7.3.1.6 for each specific safety function. The interlocks for maintenance bypasses are discussed in Subsection 7.1.3.11.

7.3.1.8 Redundancy

There are four redundant ESF trains for all ESF systems, except as specifically identified in Subsection 7.3.1.5. In addition, within each train, ESFAS and SLS controllers are redundant. Therefore, a controller failure or a controller taken out of service for maintenance has no adverse affect on the protective function. The reliability of the ESFAS/SLS, as analyzed in the PRA, is based on having two controllers in service.

7.3.1.9 Diversity

All ESF systems are automatically initiated from signals that originate in the RPS. Manual actuation of ESF systems is carried out through a diverse signal path that bypasses the RPS.

The SLS receives signals from the DAS to actuate ESF plant components. These signals are interfaced from DAS via qualified isolators within the SLS. The SLS provides priority logic to combine the DAS and SLS signals and to ensure the safety function always has priority. The DAS/SLS interface is described in MAUP-07006 (Reference 7.3-3) Sections 6.2.1.3 and 6.2.4, and shown in Figure 7.3-1.

7.3.1.10 Defense-In-Depth/Design Features

The ESFAS and SLS implement the ESF system echelon of defense-in-depth scheme, as described in Subsection 7.1.3.1.

7.3.1.11 Turbine Trip to Prevent Unnecessary Emergency Core Cooling System Actuation

The turbine is tripped on a reactor trip or high-high SG water level in any SG. Turbine trip on RT is an un-credited non-safety function in the safety analysis. However, turbine trip on RT is assumed in the safety analysis in order to prevent unnecessary ECCS actuation and to shift to the safe shutdown state by appropriate actions after AOO and PA conditions. Turbine trip on RT cannot be completely designed as Class 1E because the equipment to execute the turbine trip is located in the turbine building, which is seismic category II. Therefore, turbine trip on RT is designed as reliably as possible by applying the following design concepts:

(1) Turbine trip on reactor trip is designed as an “Associated Circuit” per IEEE Std 384-1992 (Reference 7.3-4) and RG 1.75 (Reference 7.3-5).

(2) The cables outside electrical cabinets in the turbine building are routed in dedicated raceways.

(3) Four turbine trip solenoid valves are arranged in a 1-out-of-2 configuration. A trip will be generated by train A or train D. The power for each turbine trip solenoid valve is supplied by a separate Class 1E power source (one per train).

The turbine trip signals are interfaced from the SLS, which receives RT signals from the RTBs. The design is shown in Figure 7.3-4.

7.3.1.12 Block Turbine Bypass and Cooldown Valves

There are two ESFAS trains for block turbine bypass and cooldown valves, train A and train D.

The block turbine bypass valve signal is initiated on the following conditions:

- Low-low T_{avg} signal: This signal is present when 2-out-of-4 RCS loops indicate low-low T_{avg}.
- The manual turbine bypass block switch is selected to close (manual actuation).

The resulting block turbine bypass valve signal blocks the opening of the turbine bypass valves to the condenser. This block does not affect the turbine bypass valves to the condenser, which are referred to as the “turbine bypass cooldown valves.” The block is automatically removed when the low-low T_{avg} signal goes above its setpoint.

Logic for this function is included on Figure 7.2-2 sheet 10.

The block cooldown turbine bypass signal initiates on the same low-low T_{avg} signal and manual actuation, as described above. The resulting block cooldown turbine bypass signal blocks the turbine bypass cooldown valves to the condenser. The block is automatically removed when the low-low T_{avg} signal goes above its setpoint.

The block cooldown turbine bypass signal may be manually overridden by the turbine bypass block bypass control switch (operating bypass). The override can only be actuated after the signal is actuated from low-low T_{avg}. The override is automatically reset when the low-low T_{avg} signal returns to normal. The override may be manually reset from the turbine bypass block bypass control switch. The block turbine bypass override has no effect on the block turbine bypass valve signal (i.e., it only affects the cooldown signal).

Logic for this function is included on Figure 7.2-2 sheet 10.

Block turbine bypass and cooldown valves is a non-safety function. However, block turbine bypass and cooldown valves is assumed in the safety analysis in order to prevent an excessive cooldown due to multiple valves being open. Block turbine bypass and cooldown valves cannot be completely designed as Class 1E because the equipment to execute the block turbine bypass and cooldown valves is located in the turbine building, which is seismic category II. Therefore, block turbine bypass and cooldown valves is designed as reliably as possible by applying the following design concepts:

(1) Block turbine bypass and cooldown valves is designed as an "Associated Circuit" per IEEE Std 384-1992 (Reference 7.3-4) and RG 1.75 (Reference 7.3-5).

(2) The cables outside electrical cabinets in the turbine building are routed in dedicated raceways.

Block turbine bypass and cooldown valves signal is interfaced from the SLS to the turbine bypass and cooldown valves. The design is shown in Figure 7.3-5.

7.3.2 Design Basis Information

7.3.2.1 Single Failure Criterion

The ESF systems meet the single failure criterion through multiple redundant and independent trains for all safety functions. Out of service times for ESFAS/SLS trains are limited by the technical specifications.

The potential for spurious actuation due to single failures is minimized by the use of 2-out-of-4 logic at the system level for automatic initiation functions and 2-out-of-2 logic within each train for manual actuation functions.

7.3.2.2 Quality of Components and Modules

All safety functions of the ESF systems are implemented using Class 1E components. Non-safety functions are isolated from the ESF systems with the exception of the turbine trip outputs, which are treated as associated circuits as described in Subsection 7.3.1.11.

7.3.2.3 Independence

The independence and separation within the ESFAS and SLS is as described in Subsection 7.1.3.4.

7.3.2.4 Defense-In-Depth and Diversity

Refer to Subsections 7.3.1.9 and 7.1.3.1.

7.3.2.5 System Testing and Inoperable Surveillance

Refer to Subsection 7.1.3.14.

7.3.2.6 Use of Digital Systems

All ESF systems rely on digital systems. Refer to Subsections 7.1.3.8 and 7.1.3.17.

7.3.2.7 Setpoint Determination

The safety functions performed by the ESF systems rely primarily on sensor inputs and setpoints from the RPS. In addition, setpoints for ESF systems meet the requirements of RG 1.105 (Reference 7.3-6). Refer to Subsection 7.2.2.7. The instrument accuracy, setpoint, and response time described in Table 7.3-4 are determined by applying the methodology. Technical Report MUAP-09022 (Reference 7.3-11) provides more detail description for setpoint methodology and channel uncertainty calculations for ESF functions. Setpoints determination for inputs interfaced directly into the ESFAS, such as under voltage inputs for emergency load sequencing, utilize the same setpoint determination methodology or a setpoint methodology that is specific to the instrument, as recommend by the original equipment manufacturer (OEM).

7.3.2.8 Equipment Qualification

Refer to Subsection 7.1.3.7 for details.

7.3.3 Analysis

Detailed compliance to the GDC, IEEE Std 603-1991 (Reference 7.3-7) and IEEE Std 7-4.3.2-2003 (Reference 7.3-8) is described in MUAP-07004 Section 3.0, Appendix A and B.

7.3.3.1 FMEA

The FMEA method for ESF actuation in the PSMS is identical to that used in the RPS, as described in Subsection 7.2.3.1. The safe state for ESFAS/SLS is “as-is.”

The FMEA for ESF actuation in PSMS is provided in Table 7.3-7. Figure 7.3-~~5~~6 shows the configuration of system diagram for ESFAS as used in the FMEA table. Train A is illustrated as the representative features in this figure, while the PSMS consists of four trains.

7.3.3.2 Safety Analysis

The ESF system design requirements such as response time and setpoint determination, are considered and reflected in the safety analysis, contained in Chapter 15. The response time, instrument accuracy, and setpoint as shown in Table 7.3-4, meet the safety analysis assumptions.

Chapter 15 analysis for US-APWR addresses AOOs and PAs.

Table 7.3-3 Engineered Safety Features Actuation Signals
(Sheet 3 of 3)

Actuation Signal	Number of Sensors, Switches, or Signals	Actuation Logic	Permissives and Bypasses
			For Permissives and Bypasses Refer Table 7.2-4
~~12. Block Turbine Bypass valves - Logic Diagram Figure 7.2-2 Sheet 10~~			
~~Low-Low T_{avg} Signal~~	~~4 Temperature sensors (Shared with RT)~~	~~2/4 for train~~	~~Manually bypassed from turbine bypass block bypass switch for cooldown turbine bypass valves and turbine bypass valve.~~
~~Turbine Bypass Block Bypass Switch~~	~~1 Switch per train~~	~~1/1 per train~~	~~None~~

Table 7.3-4 Engineered Safety Features Actuation Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal)
(Sheet 2 of 2)

ESF Function	Variables to be monitored	Range of Variables	Instrument Accuracy*[1,2]	Response Time*[1,2,3]	Setpoint*[4]
Emergency Feedwater Isolation					
(a)High SG Water Level	SG Water Level	0 to 100% of span (narrow Range taps)	3% of span	3.0 sec	50% of span
(b)Low Main Steam Line Pressure	Main Steam Line Pressure	0 to 1400 psig	3% of span	3.0 sec	525 psig
CVCS Isolation					
High Pressurizer Water Level	Pressurizer Water Level	0 to 100% of span	3% of span	3.0 sec	92% of span
~~**Block Turbine Bypass and Cooldown Valves**~~					
~~Low-Low T_{avg}~~	~~Reactor Coolant Temperature~~	~~-~~	~~2.0°F~~	~~8.0 sec~~	~~553°F~~

Note:
1. Instrument accuracy and response time calculation methodology refer to Subsection 7.2.2.7.
2. Instrument accuracies and response times will be decided to take into account the specification of instruments.
3. Additional time during LOOP is referred to Chapter 8.
4. Setpoints will be adjusted to compensate for loop accuracy.

Table 7.3-6 ESF Actuation System - Manual Reset and Bypass (Software Switches) (Sheet 1 of 2)

Manual Control*[1]	Trains				Fig 7.2-2
Manual Reset for EFW Actuation Train A	A				Sheet 7
Manual Reset for EFW Actuation Train B		B			Sheet 7
Manual Reset for EFW Actuation Train C			C		Sheet 7
Manual Reset for EFW Actuation Train D				D	Sheet 7
Manual Bypass Control for EFW Isolation Train A	A				Sheet 8
Manual Bypass Control for EFW Isolation Train B		B			Sheet 8
Manual Bypass Control for EFW Isolation Train C			C		Sheet 8
Manual Bypass Control for EFW Isolation Train D				D	Sheet 8
Manual Reset for Main Steam Line Isolation #1	A				Sheet 9
Manual Reset for Main Steam Line Isolation #2				D	Sheet 9
Manual Reset & Block ECCS Actuation Train A*[2]	A				Sheet 11
Manual Reset & Block ECCS Actuation Train B*[2]		B			Sheet 11
Manual Reset & Block ECCS Actuation Train C*[2]			C		Sheet 11
Manual Reset & Block ECCS Actuation Train D*[2]				D	Sheet 11
Manual Bypass Control for ECCS Train A on Low Pressurizer Pressure	A				Sheet 11
Manual Bypass Control for ECCS Train B on Low Pressurizer Pressure		B			Sheet 11
Manual Bypass Control for ECCS Train C on Low Pressurizer Pressure			C		Sheet 11
Manual Bypass Control for ECCS Train D on Low Pressurizer Pressure				D	Sheet 11
Manual Bypass Control for CVCS Isolation Train A	A				Sheet 6
Manual Bypass Control for CVCS Isolation Train B		B			Sheet 6
Manual Bypass Control for CVCS Isolation Train C			C		Sheet 6
Manual Bypass Control for CVCS Isolation Train D				D	Sheet 6
Manual Reset for CVCS Isolation #1	A				Sheet 6
Manual Reset for CVCS Isolation #2				D	Sheet 6
Manual Reset for EFW Isolation of Loop A (EFW Control Valve)	A				Sheet 8
Manual Reset for EFW Isolation of Loop B (EFW Control Valve)		B			Sheet 8
Manual Reset for EFW Isolation of Loop C (EFW Control Valve)			C		Sheet 8
Manual Reset for EFW Isolation of Loop D (EFW Control Valve)				D	Sheet 8
Manual Reset for EFW Isolation of Loop A (EFW Isolation Valve)		B			Sheet 8
Manual Reset for EFW Isolation of Loop B (EFW Isolation Valve)	A				Sheet 8
Manual Reset for EFW Isolation of Loop C (EFW Isolation Valve)				D	Sheet 8
Manual Reset for EFW Isolation of Loop D (EFW Isolation Valve)			C		Sheet 8
Manual Bypass Control for ECCS Actuation Train A on Low Main Steam Line Pressure	A				Sheet 9
Manual Bypass Control for ECCS Actuation Train B on Low Main Steam Line Pressure		B			Sheet 9
Manual Bypass Control for ECCS Actuation Train C on Low Main Steam Line Pressure			C		Sheet 9
Manual Bypass Control for ECCS Actuation Train D on Low Main Steam Line Pressure				D	Sheet 9
Manual Bypass Control for MFW Isolation Train A	A				Sheet 10
Manual Bypass Control for MFW Isolation Train B		B			Sheet 10
Manual Bypass Control for MFW Isolation Train C			C		Sheet 10
Manual Bypass Control for MFW Isolation Train D				D	Sheet 10
~~Manual Bypass Control for Turbine Bypass Block Train A~~	~~A~~				~~Sheet 10~~
~~Manual Bypass Control for Turbine Bypass Block Train B~~		~~B~~			~~Sheet 10~~
~~Manual Bypass Control for Turbine Bypass Block Train C~~			~~C~~		~~Sheet 10~~
~~Manual Bypass Control for Turbine Bypass Block Train D~~				~~D~~	~~Sheet 10~~

Table 7.3-6 ESF Actuation System - Manual Reset and Bypass (Software Switches) (Sheet 2 of 2)

Manual Control*[1]	Trains				Fig 7.2-2
Manual Reset for MFW Regulation Valve Closure #1	A				Sheet 10
Manual Reset for MFW Regulation Valve Closure #2				D	Sheet 10
Manual Reset for MFW Isolation #1	A				Sheet 10
Manual Reset for MFW Isolation #2				D	Sheet 10
Manual Reset for CS Actuation Train A	A				Sheet 12
Manual Reset for CS Actuation Train B		B			Sheet 12
Manual Reset for CS Actuation Train C			C		Sheet 12
Manual Reset for CS Actuation Train D				D	Sheet 12
Manual Reset for Containment Isolation Phase B Train A	A				Sheet 12
Manual Reset for Containment Isolation Phase B Train B		B			Sheet 12
Manual Reset for Containment Isolation Phase B Train C			C		Sheet 12
Manual Reset for Containment Isolation Phase B Train D				D	Sheet 12
Manual Reset for Containment Isolation Phase A #1	A				Sheet 12
Manual Reset for Containment Isolation Phase A #2				D	Sheet 12
Manual Reset for Containment Purge Isolation #1	A				Sheet 12
Manual Reset for Containment Purge Isolation #2				D	Sheet 12
Manual Reset for MCR Isolation Train A	A				Sheet 12
Manual Reset for MCR Isolation Train B		B			Sheet 12
Manual Reset for MCR Isolation Train C			C		Sheet 12
Manual Reset for MCR Isolation Train D				D	Sheet 12
~~Manual Turbine Bypass Block Actuation Train A~~	~~A~~				~~Sheet 10~~
~~Manual Turbine Bypass Block Actuation Train B~~		~~B~~			~~Sheet 10~~
~~Manual Turbine Bypass Block Actuation Train C~~			~~C~~		~~Sheet 10~~
~~Manual Turbine Bypass Block Actuation Train D~~				~~D~~	~~Sheet 10~~

Note:
1. Manual controls are located in the MCR unless otherwise noted.
2. Block permitted with active P-4
3. All bypass controls (operating bypasses) permitted with active P-11, and automatically unbypassed by inactive P-11.

Table 7.3-7 FMEA for ESF Actuation in PSMS (for Figure 7.3-~~5~~6) (Sheet 1 of 3)

Component (one train)*[1]	Failure Mode	Method of Failure Detection	Local Failure Effect	Effect on Protective Function
Sensor	Fail high	Self-diagnostic alarm from the affected RPS train. Annunciation of partial actuation from the affected RPS train. Cross channel comparison.	Bistable changes to actuation state and partial actuation signal is generated in the affected RPS train.	ESF actuation logic becomes 1-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation. If unrestricted bypass of one instrument channel has already been executed in another train, ESF actuation logic becomes 1-out-of-2 due to the sensor failure. Remaining two trains provide ESF actuation.
	Fail low	Self-diagnostic alarm from the affected RPS train. Annunciation of partial actuation from the affected RPS train. Cross channel comparison.	Bistable changes to actuation state and partial actuation signal is generated in the affected RPS train.	ESF actuation logic becomes 1-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation. If unrestricted bypass of one instrument channel has already been executed in another train, ESF actuation logic becomes 1-out-of-2 due to the sensor failure. Remaining two trains provide ESF actuation.
	Fail as is	Cross channel comparison.	Bistable does not change to actuation state in the affected RPS train when process reaches actuation level.	ESF actuation logic becomes 2-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation.
RPS Input part (from Sensor)	Fail high	Self-diagnostic alarm from the affected RPS train. Annunciation of partial actuation from the affected RPS train. Cross channel comparison.	Bistable changes to actuation state and partial actuation signal is generated in the affected RPS train.	ESF actuation logic becomes 1-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation. If unrestricted bypass of one instrument channel has already been executed in another train, ESF actuation logic becomes 1-out-of-2 due to the sensor failure. Remaining two trains provide ESF actuation.
	Fail low	Self-diagnostic alarm from the affected RPS train. Annunciation of partial actuation from the affected RPS train. Cross channel comparison.	Bistable changes to actuation state and partial actuation signal is generated in the affected RPS train.	ESF actuation logic becomes 1-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation.
	Fail as is	Cross channel comparison.	Bistable does not change to actuation state in the affected RPS train when process reaches actuation level.	ESF actuation logic becomes 2-out-of-3 due to the sensor failure. Remaining three trains provide ESF actuation.

Table 7.3-7 FMEA for ESF Actuation in PSMS (for Figure 7.3-~~5~~6) (Sheet 2 of 3)

Component (one train)*1	Failure Mode	Method of Failure Detection	Local Failure Effect	Effect on Protective Function
RPS Processing part (in RPS)	No data output	Self-diagnostic alarm from the affected RPS train. Annunciation of communication error from other RPS trains.	Partial actuation signal does not reach to other RPS trains when process reaches actuation level.	ESF actuation logic becomes 2-out-of-3 due to the processing failure. Remaining three trains provide ESF actuation.
	Spurious Status Change	Ractor Trip Annunciation of the affected RPS train	The affected RPS train emits Reactor Trip Signal.	Ractor Trip logic becomes 1-out-of-3 due to the spurious signal. Remaining three trains provide ESF actuation.
RPS Communication part (between RPS trains)	No data output	Annunciation of communication error from the affected other RPS trains.	Partial actuation signal does not reach to other RPS trains when process reaches actuation level. Trip signals from other RPS trains dose not reach to the affected train when process reaches actuation level.	ESF actuation logic becomes 2-out-of-3 due to the communication failure. Remaining three trains provide ESF actuation.
RPS Communication part (to ESFAS)	No data output	Annunciation of communication error from the affected ESFAS trains.	Partial actuation signal does not reach to the ESFAS trains when process reaches actuation level.	ESF actuation logic becomes 2-out-of-3 due to the communication failure. Remaining three trains provide ESF actuation.
ESFAS Communication part (from RPS)	No data output	Self-diagnostic alarm in the affected ESFAS train.	One parallel redundant part fails. Another parallel redundant part process the signal.	No failure effect to ESF function.
ESFAS Processing part (in ESFAS)	No data output	Self-diagnostic alarm in the affected ESFAS train.	One parallel redundant part fails. Another parallel redundant part process the signal.	No failure effect to ESF function.
	Spurious Status Change	ESF Actuation Annunciation of the affected ESFAS train	The affected train actuates ESF function.	Remainig three trains provide appropriate ESF actuation.
ESFAS Communication part (to SLS)	No data output	Self-diagnostic alarm in the affected SLS train.	One parallel redundant part fails. Change to another parallel redundant part.	No failure effect to ESF function.

Table 7.3-7 FMEA for ESF Actuation in PSMS (for Figure 7.3-~~5~~6) (Sheet 3 of 3)

Component (one train)*[1]	Failure Mode	Method of Failure Detection	Local Failure Effect	Effect on Protective Function
Safety Bus	No data input or output	Self-diagnostic alarm in the affected ESFAS and SLS train.	One train ESF does not actuate when process reaches actuation level.	Three trains of ESF can be actuated due to the communication failure. Remaining three trains provide ESF actuation.
	Fail to dis-connection	Self-diagnostic alarm in the affected ESFAS and SLS train.	There is no impact for a single disconnection due to its ring configuration of the safety bus.	No failure effect to ESF function.
SLS Communication part (from ESFAS)	No data output	Self-diagnostic alarm in the affected SLS train.	One parallel redundant part fails. Another parallel redundant part process the signal.	No failure effect to ESF function.
SLS Processing part (in SLS)	No data output	Self-diagnostic alarm.	One parallel redundant part fails. Another parallel redundant part process the signal.	No failure effect to ESF function.
	Spurious Status Change	Refer to MUAP-09020 "~~FMEA of~~Functional Assignment Analysis for Safety I&C System"	Refer to MUAP-09020 "~~FMEA of~~ Functional Assignment Analysis for Safety I&C System"	Refer to MUAP-09020 "~~FMEA of~~ Functional Assignment Analysis for Safety I&C System"
SLS Output part (to Component)	Spurious change status	Manual periodic test or plant system disturbance.	One train ESF changes their status. (If the change of their status affects plant disturbances, appropriate design such as duplicated output module are adopted.)	One train of ESF can be actuated due to the output failure. All trains can still provide ESF actuation. The periodic test is administrated to detect the failure for components whose spurious actuation does not cause a plant disturbance.
	Fail as is	Manual periodic test.	One train ESF does not actuate when process reaches actuation level.	One train of ESF can fail due to the output failure. The remaining one or three trains provide ESF actuation, depending on the two or four train mechanical system configuration. For four train mechanical systems, if another train is being tested, two trains provide ESF actuation. The periodic test is administrated to detect the failure.
Unit Bus Inside part of PSMS	No data output	Annunciation of communication error	No failure effect to ESF function.	No failure effect to ESF function.
Unit Bus Outside part of PSMS	Fail to dis-connection	Annunciation of communication error	There is no impact for a single disconnection due to its ring configuration of the safety bus.	No failure effect to ESF function.

Note:

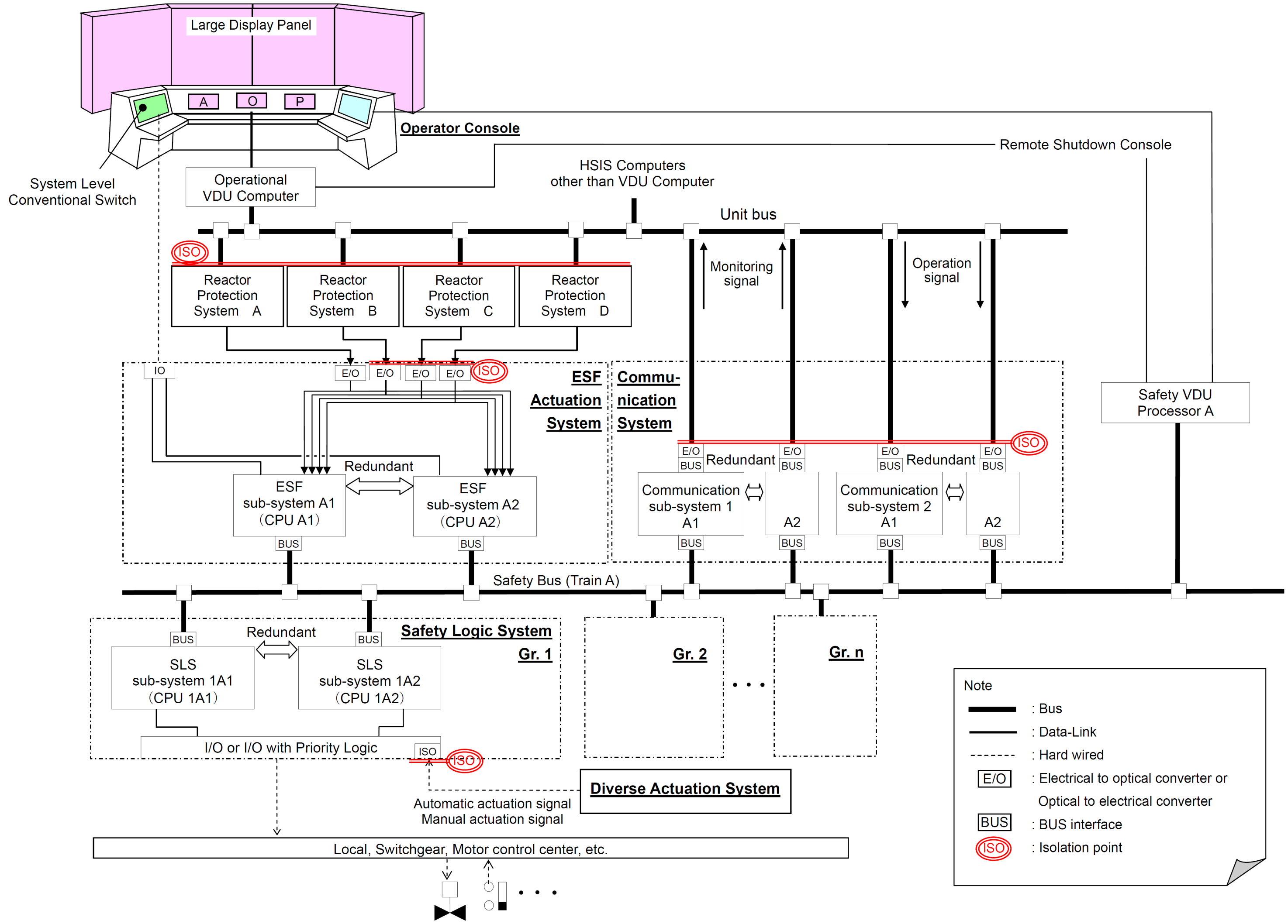


Figure 7.3-1 Configuration of Engineered Safety Features Actuation System and Safety Logic System

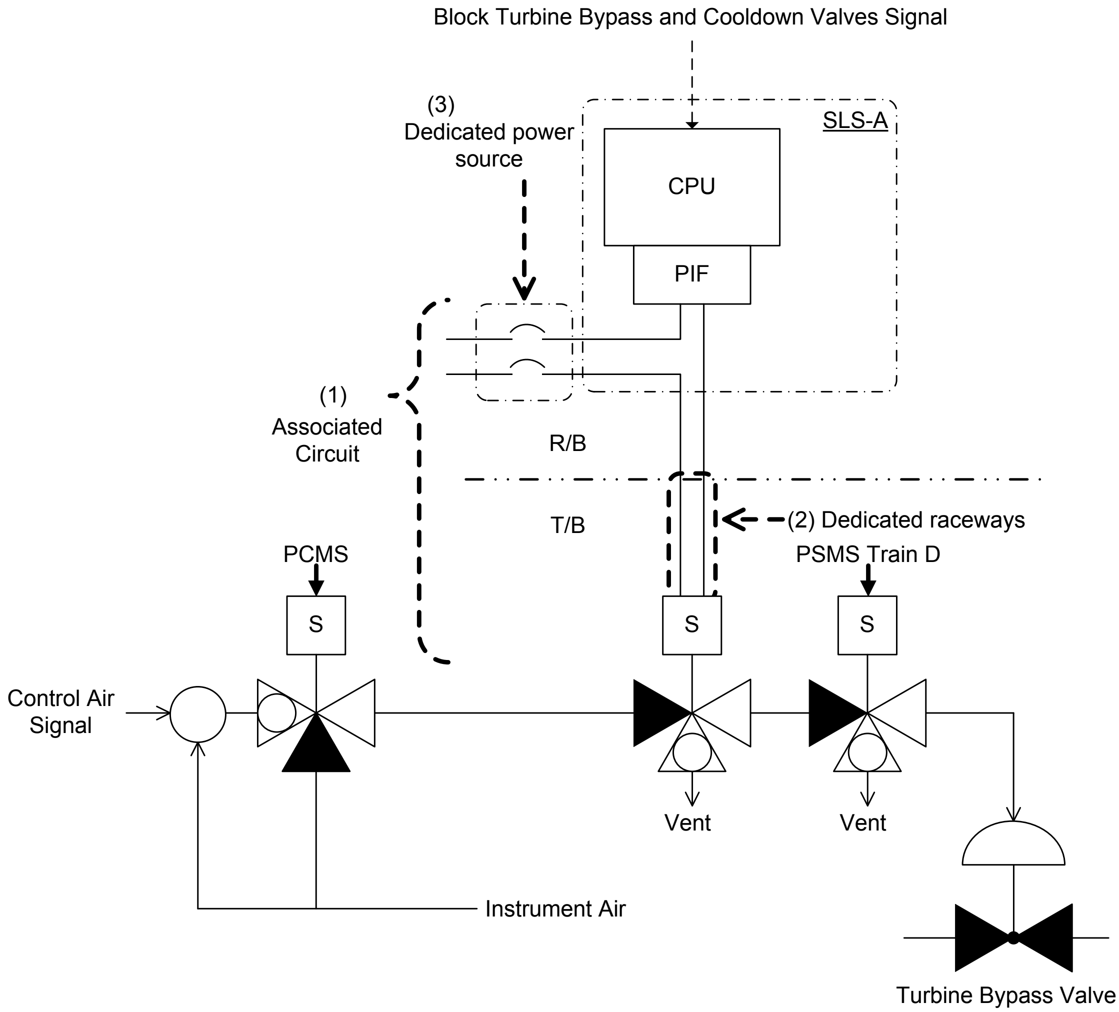
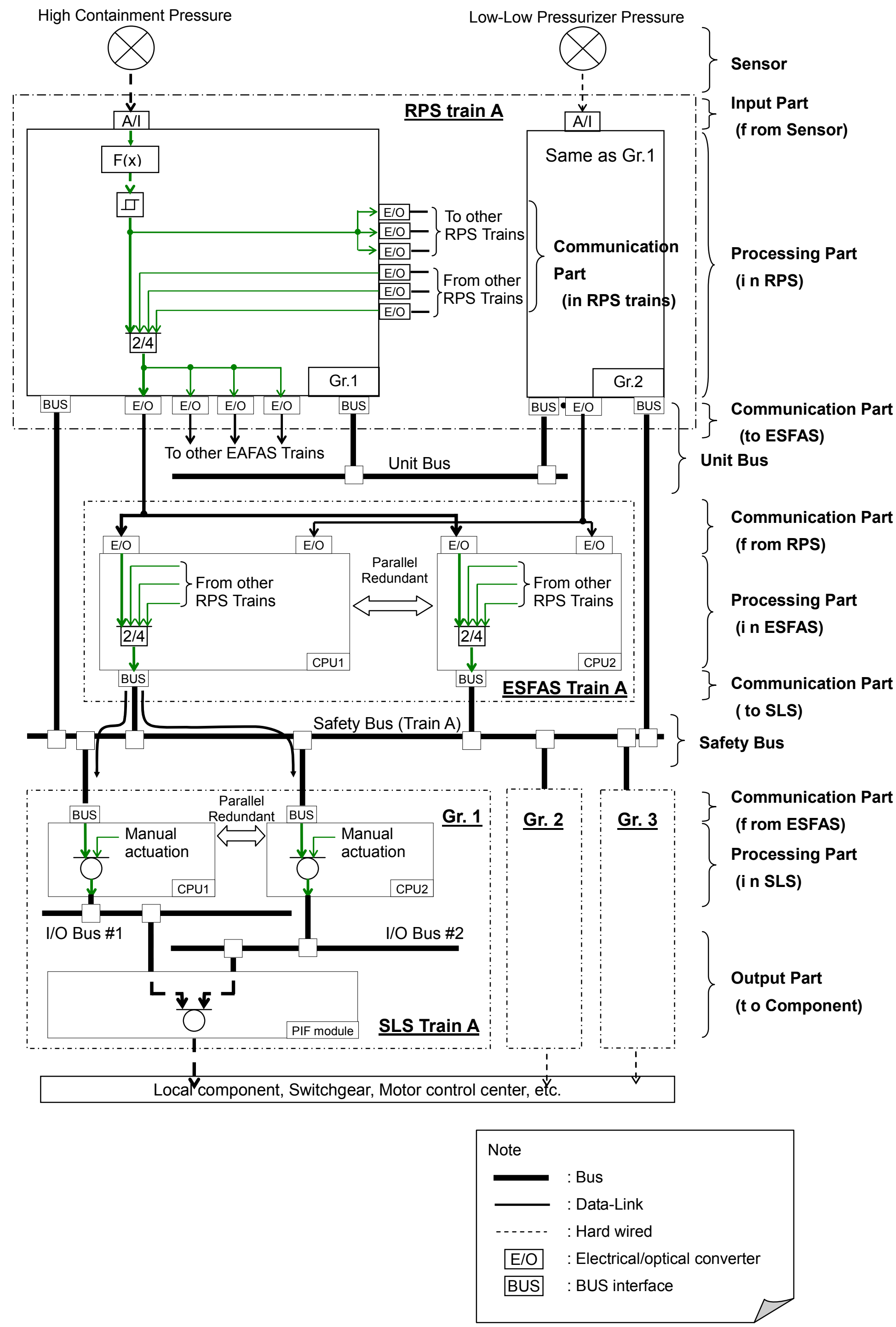


Figure 7.3-5 Summary of Design Concept for Block Turbine Bypass and Cooldown Valves



(Train A is shown for representative train.)

Figure 7.3-6~~5~~ Configuration of ESF System for Use in FMEA (for Table 7.3-7)

diversity features, single failure criterion, quality of components and modules, independence, periodic testing, and use of digital systems.

7.4.1.4 HSIS

All functions needed to achieve and maintain both normal and safe shutdown can be manually initiated and monitored by operators using the operational VDUs or the safety grade HSIS. The operational VDUs ~~can~~ provide HSI for all safety and non safety-related safe shutdown functions. The safety grade HSIS provides all safety-related controls and plant information, including critical parameters required for post accident conditions. The operational VDUs and safety grade HSIS are accessible in the MCR and the RSR.

Tables 7.4-1 and 7.4-2 provide a list of component controls and instrumentation used to achieve safe shutdown.

7.4.1.5 Normal and Safe Shutdown from Outside the MCR

GDC 19 (Reference 7.4-5) requires, "Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot standby of the reactor, including necessary I&C systems to maintain the unit in a safe condition during hot standby, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures."

In the event the MCR is uninhabitable for any reasons including fire, the control and monitoring of normal and safe shutdown functions can be performed from the RSR, which is located outside the MCR fire area in the reactor building. This capability meets the requirements of GDC 19.

The requirements for designing the RSR are: (1) LOOP is possible following the evacuation from MCR, and (2) during normal operation, operators may have to evacuate the MCR immediately, without any action to plant, whenever they decide that evacuation from the MCR is necessary.

The RSR is designed in accordance with the following principles based on the above requirements.

1. The RSR is designed to shutdown the reactor, maintain the reactor in hot standby condition, and transition the reactor safely to cold shutdown. There are no unique required control actions outside the RSR to achieve or maintain hot standby or cold shutdown. Periodic RCS effluent sampling is a local operation for shutdown from the RSR, as it is for shutdown from the MCR.

2. The I&C equipment in the RSR is electrically isolated from any credible faults that may originate in the MCR. In addition, I&C equipment in the RSR is not affected by any spurious signals that may originate in the MCR. Prior to activation of HSI at the RSR, it is assumed that there are no prior failures that adversely affect the operability of I&C equipment in the RSR.

7.4.1.6 Normal and Safe Shutdown Functions

HSI is provided in the MCR and RSR for control of normal and safe shutdown plant components and for monitoring functions as shown in Tables 7.4-1 and 7.4-2, respectively. Shutdown functions consist of normal shutdown operation, and safe shutdown operation (i.e., safe shutdown using only safety-related plant equipment). These shutdown functions are described as follows.

The COL applicant is to provide a description of component controls and indications required for safe shutdown related to the ultimate heat sink (UHS).

7.4.1.6.1 Normal Shutdown

7.4.1.6.1.1 Hot Standby

The primary functions and related process systems (shown in parenthesis) required to achieve and maintain hot standby are as follows.

(1) Shutdown the reactor using control rods.

(2) Supply boric acid water to RCS for shutdown (CVCS).

(3) Remove heat of RCS by the following measures:

  (i) Main steam release by turbine bypass system or to the atmosphere (main steam supply system [MSS]).

  (ii) Provide feedwater to SGs (condensate and feedwater system [CFS] and MSS).

(4) Control pressure of the RCS (RCS and CVCS).

(5) Supply instrument air (instrument air system [IAS]).

(6) Assure CCW and ESW (CCWS and ESWS).

(7) Provide HVAC function to the required areas including the containment, MCR~~.~~ (HVAC).

(8) Utilize power systems, which support the above functions, for LOOP.

7.4.1.6.1.2 Hot and Cold Shutdown

The primary functions and related process systems (shown in parenthesis) required to achieve and maintain cold shutdown are as follows. This describes functions to achieve and maintain cold shutdown from hot standby, therefore this includes functions to achieve and maintain hot shutdown. The capabilities and limitations of these systems are defined in the sections of this document that describe the respective process systems.

(1) Remove heat from the RCS by the following measures:

(i) Main steam release by turbine bypass system or to the atmosphere (MSS).

(ii) Provide feedwater to SGs (CFS and MSS).

(iii) Use RHRS (RHRS).

(2) Control pressure and inventory of RCS (RCS and CVCS).

(3) Supply boric acid water to RCS (CVCS).

(4) Sample the boron concentration in RCS (process and post-accident sampling system [PSS]).

(5) Supply instrument air (IAS).

(6) Assure CCW and ESW (CCWS and ESWS).

(7) Provide HVAC function to the required areas including the containment, MCR (HVAC).

(8) Monitor neutron flux.

(9) Manually initiate appropriate ESF system shutdown operating bypasses.

7.4.1.6.2 Safe Shutdown

7.4.1.6.2.1 Hot Standby

The primary functions and related process systems (shown in parenthesis) required to achieve and maintain hot standby using only safety-related equipment are as follows.

(1) Trip the reactor, which accomplishes the reactor shutdown condition.

(2) Remove heat from RCS by the following measures:

(i) Main steam release to the atmosphere (MSS).

(ii) Provide EFW to SGs (EFWS and MSS).

(3) Control pressure of the RCS (RCS).

(4) Provide HVAC functions to the required areas including the MCR (HVAC).

(5) Utilize the emergency power system for the above functions in the event of LOOP.

7.4.1.6.2.2 Hot and Cold Shutdown

The primary functions and related process systems (shown in parenthesis) required to achieve and maintain cold shutdown using only safety-related equipment are as follows. This describes functions to achieve and maintain cold shutdown from hot standby, therefore this includes functions to achieve and maintain hot shutdown. The capabilities and limitations of these systems are defined in the sections of this document that describe the respective process systems.

(1) Remove heat of RCS by the following measures:

(i) Main steam release to the atmosphere (MSS).

(ii) Provide EFW to SGs (EFWS and MSS).

(iii) Use RHRS (RHRS).

(2) Control pressure and inventory of RCS (RCS).

(3) Supply boric acid water to RCS (safety injection system [SIS]).

(4) Assure CCW and ESW (CCWS and ESWS).

(5) Provide HVAC function to the required areas including the MCR (HVAC).

(6) Monitor the neutron flux.

(7) Manually initiate appropriate ESF system shutdown operating bypasses.

(8) Utilize the emergency power system for the above functions in the event of LOOP.

7.4.2 Design Basis Information

The US-APWR normal and safe shutdown design, including the design of the RSR, is based on the following codes and standards:

1. 10 CFR 50, Appendix A General Design Criteria 19 “Control Room.”

2. RG 1.68.2 (Reference 7.4-7), Rev. 1, “Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-cooled Nuclear Power Plants.”

3. Standard Review Plan 7.4, Rev. 5, “Safe Shutdown System.”

4. RG 1.189 (Reference 7.4-8), “Fire Protection for Operating Nuclear Power Plants.”

In addition, the design of the safe shutdown systems including the design of the RSR is based on the following CFR;

1. 10 CFR 50.55a(a)(1), “Quality Standards.”
2. 10 CFR 50.55a(h), “Protection Systems and Safety Systems”
3. 10 CFR 50.34(f)(2)(xx), “Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves”
4. 10 CFR 50, Appendix A, GDC 1, “Quality Standards and Records.”
5. GDC 2, “Design Bases for Protection against Natural Phenomena.”
6. GDC 4, “Environmental and Missile Design Bases.”
7. GDC 13, “Instrumentation and Control.”
8. GDC 24, “Separation of Protection and Control Systems.”
9. GDC 34, “Residual Heat Removal.”
10. GDC 35, “Emergency Core Cooling.”
11. GDC 38, “Containment Heat Removal.”

7.4.2.1 I&C Systems Required for Safe Shutdown

Safe shutdown, using only safety-related equipment, relies on monitoring instrumentation interfaced through the PSMS and safety grade HSI.

7.4.2.2 Single Failure Criterion

All functions of the RPS, ESFAS, SLS, and safety grade HSIS, including those used to achieve safe shutdown, meet the single failure criterion. Safety-related plant instrumentation and component controls, used to achieve safe shutdown, are redundant.

The PSMS is the I&C system credited for safe shutdown. The PSMS meets the single failure criteria through multiple redundant and independent trains, which control multiple redundant and independent mechanical trains, as described in ~~Topical Report,~~ MUAP-07004 Section 4.1(b). The PSMS is completely isolated from all non-safety I&C systems, such that there are no failures in non-safety I&C systems that can adversely affect the PSMS, as described in ~~Topical Report,~~ MUAP-07004 Section 3.3(6).

The test method for all I&C equipment within the PSMS, including equipment used for safe shutdown, is the same. Self-diagnosis with overlapping manual tests that encompass PSMS I/O and interfacing plant process components, such as sensors, pumps and valves, ensure there are no undetectable failures. There are at least two fully redundant and independent trains for all safe shutdown components to satisfy the single failure criterion. Table 7.4-1 shows that there is redundancy for each component credited for safe shutdown.

7.4.2.3 Quality of Components and Modules

All functions of the RPS, ESFAS, SLS, and safety grade HSIS, including those used to achieve safe shutdown, are Class 1E, and meet all appropriate quality requirements. Class 1E plant instrumentation and component controls are provided for all safe shutdown functions.

The operational VDUs and interfaces to the SLS, which may also be used to achieve normal and safe shutdown, are developed through an augmented quality program that includes software V&V, and seismic and environmental testing to levels consistent with the PSMS.

7.4.2.4 Independence

Redundant divisions of the RPS, ESFAS, SLS, and safety grade HSI, including those used to achieve safe shutdown, are independent from each other and from the non-safety division. This independence is also applicable to redundant divisions of safety-related plant instrumentation and component controls for all safe shutdown functions as described in Subsections 7.1.3.4 and 7.1.3.5.

Within the PSMS, which is the I&C system credited for safe shutdown, there are no components that are common to redundant trains, such as common switches for actuation, reset, mode, test, or any other features which could compromise the independence of the redundant trains.

Within the mechanical systems credited for safe shutdown, the main steam isolation valves and main feedwater isolation valves are common to both redundant safe shutdown trains. Each valve has two separate and redundant solenoid operators which are assigned to separate trains.

7.4.2.5 Periodic Testing

All functions of the RPS, ESFAS, SLS, and safety grade HSI, including those used to achieve safe shutdown, are periodically tested, as described in Subsection 7.1.3.14. This testing encompasses safety-related plant instrumentation and component controls for all safe shutdown functions. It is noted that fast response RTDs are not used for the RTDs of wide range RCS temperature for safe shutdown and PAM, therefore, the response time testing in BTP 7-13 (Reference 7.4-9) is not applicable to the RTDs in this section.

7.4.2.6 Use of Digital Systems

All functions of the PCMS, used to achieve normal shutdown, and all functions of the RPS, ESFAS, SLS, and safety grade HSI, including those used to achieve safe shutdown, rely on digital systems, as described in Subsections 7.1.3.8 and 7.1.3.17. Analog plant instrumentation and conventional electro-mechanical component (e.g., solenoids, motor starters and switchgears) ~~controls~~ are relied on for ~~normal and~~ safe shutdown functions.

7.4.3 Analysis

Detailed compliance to the GDC, IEEE Std 603-1991 and IEEE Std 7-4.3.2-2003 are described in MUAP-07004 Section 3.0, Appendix A and B.

Table 7.4-1 Component Controls for Shutdown (Sheet 1 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
RT System	RTB	No	Yes	2	4	
RCS	RCP	Yes	No	-	-	Available with off-site power.
	Safety Depressurization Valve	No	Yes	1	2	Note1
	Safety Depressurization Valve Block Valve	No	Yes	1	2	Note1
	Pressurizer Heater Backup Group	No	Yes	2	4	
	Pressurizer Spray Valve	Yes	No	-	-	
	Reactor Vessel (RV) Vent Valve	No	Yes	1	2	These valves could be used only if the venting becomes necessary.
CVCS	Charging Pump	Yes	No	-	-	Automatic start in LOOP.
	Charging Flow Control Valve	Yes	No	-	-	
	Letdown Line 1st (2nd) Stop Valve	Yes	No	-	-	
	Letdown Line inside C/V Isolation Valve	Yes	No	-	-	
	CHP Inlet Line VCT Side 1st, 2nd Isolation Valve	Yes	No	-	-	
	CHP Inlet Line BAT Side Isolation Valve	Yes	No	-	-	
	CHP Inlet Line RWSAT Side Isolation Valve	No	No	-	-	These valves are automatically opened on Low Volume Control Tank Water Level.

Note1: The configuration of the Safety Depressurization Valves and Safety Depressurization Valve – Block Valves meets the single failure criteria (for both electrical and mechanical failures), to ensure the capability for depressurization when required and to prevent spurious depressurization. There are two depressurization lines, each with one Safety Depressurization Valve (normally closed) and one Safety Depressurization Valve – Block Valve (normally open), each assigned to different trains. Four trains are used, such that the four valves in the two depressurization lines do not share any common train assignments. Should a Safety Depressurization valve fail to open when required, depressurization can be achieved through the other line. Should a Safety Depressurization valve spuriously open, the series block valve can be closed.

Table 7.4-1 Component Controls for Shutdown (Sheet 2 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
CVCS (conti-nued)	Pressurizer Auxiliary Spray Valve	Yes	No	-	-	
	RHR Letdown Line Flow Control Valve	Yes	No	-	-	
	Seal Water Return Line 1st, 2nd Isolation Valve	Yes	Yes	1	2	These valves are used to holdup seal water inside containment in Safe Shutdown.
SIS	Safety Injection Pump (SIP)	No	Yes	2	4	Table 6.3-6
	SIPs Suction Isolation Valve	No	Yes	2	4	Table 6.3-6
	SIPs Discharge Containment Isolation Valve	No	Yes	2	4	Table 6.3-6
	Direct Vessel Safety Injection Line Valve	No	Yes	2	4	Table 6.3-6
	Emergency Letdown Line 1st, 2nd Isolation Valve	No	Yes	1	2	Table 6.3-6
	Accumulator Discharge Valve	Yes	Yes	4	4	Table 6.3-6
	ACC Nitrogen Supply Line Isolation Valve	No	Yes	4	4	These valves are used in case of ACC discharge valve failure to close. Table 6.3-6
	ACC Nitrogen Discharge Valve	No	Yes	1	2	
RHRS	CS/RHR Pump	Yes	Yes	2	4	Table 5.4.7-1
	1st/2nd CS/RHR Pump Hot Leg Isolation Valve	Yes	Yes	2	4	Table 5.4.7-1
	CS/RHR Hx Outlet Flow Control Valve	Yes	No	-	-	
	CS/RHR Hx Bypass Flow Control Valve	Yes	No	-	-	
	CS/RHR Pumps RWSP Suction Isolation Valve	Yes	Yes	2	4	CSS Valves Table 6.2.2-3
	RHR Discharge Line Containment Isolation Valve	Yes	Yes	2	4	Table 5.4.7-1

Table 7.4-1 Component Controls for Shutdown (Sheet 3 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
RHRS (conti-nued	RHR Flow Control Valve	Yes	Yes	2	4	Table 5.4.7-1
	CS/RHR Pump Full-Flow Test Line Stop Valve	No	Yes	2	4	Table 5.4.7-1
EFWS	EFW Pump (Motor-Driven or Turbine Driven)	No	Yes	2	4	Table 5.4.7-1
	EFW Control Valve	No	Yes	2	4	Table 10.4.9-3
	EFW Isolation Valve	No	Yes	~~2 per 2 SG~~2	~~4 per 4 SG~~4	~~2 electrical train assigned per SG~~ Table 10.4.9-3
	T/D-EFW Pump MS Line Steam Isolation Valve	No	Yes	1~~ per pump~~	~~2 per pump~~4	Table 10.4.9-3
	T/D-EFW Pump Actuation Valve	No	Yes	1	~~2~~4	Table 10.4.9-3
MSS	Main Steam Depressurization Valve	No	Yes	2	4	Table 10.3.3-1
	Main Steam Relief Valve	Yes	No	-	-	
	Main Steam Relief Valve Block Valve	No	Yes	2	4	Table 10.3.3-1
	Main Steam Isolation Valve	Yes	Yes	4	4	Table 10.3.3-1
	Main Steam Bypass Isolation Valve	Yes	Yes	4	4	Table 10.3.3-1
	Turbine Bypass Valve	Yes	No	-	-	
CFS	MFW Bypass Regulation valve	Yes	No	-	-	
	SG Water Filling Control Valve	Yes	No	-	-	
CCWS	CCW Pump	Yes	Yes	2	4	Automatic start in LOOP. Table 9.2.2-3
	CS/RHR Hx CCW Outlet Valve	Yes	Yes	2	4	Table 9.2.2-3
ESWS	ESW Pump	Yes	Yes	2	4	Automatic start in LOOP. Table 9.2.2-3
	ESW Pump Discharge Valve	Yes	Yes	2	4	Table 9.2.2-3

Table 7.4-1 Component Controls for Shutdown (Sheet 4 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
IAS	Instrument Air Compressor	Yes	No	-	-	Automatic start in LOOP.
PSS	Letdown Demineralizer Inlet Sampling Valve	Yes	No	-	-	Local Manual Valve
	RHR Loop Sampling Stop Valve	Yes	No	-	-	Installed inside sampling rack.
	Inside Sampling Hood Isolation Valve	Yes	No	-	-	Installed inside sampling rack.
	Loop Sampling Line In and out side C/V Isolation Valve	Yes	No	-	-	
SGBDS	SGBD Line Containment Isolation Valve	No	Yes	4	4	Close on EFW Pump Start Signal. Table 10.3.3-1
	SGBD Line Isolation Valve	No	Yes	4	4	Close on EFW Pump Start Signal. Table 10.3.3-1
	SGBD Sampling Line Containment Isolation Valve	No	Yes	4	4	Close on EFW Pump Start Signal. Table 10.3.3-1
Other	ECCS Actuation Signal Block	Yes	Yes	4	4	
	Main Steam Line Pressure Signal Block	Yes	Yes	4	4	
	Emergency Power Generator	No	Yes	2	4	Automatic start in LOOP.

Table 7.4-1 Component Controls for Shutdown
(Sheet 5 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
HVAC	MCR Air Handling Unit & Damper	Yes	Yes	2	4	Automatic start in LOOP.
	Class 1E Electrical Room Air Handling Unit & Damper	Yes	Yes	2	4	Automatic start in LOOP.
	Class 1E Electrical Room Return Air Fan	Yes	Yes	2	4	Automatic start in LOOP
	Class 1E Battery Room Exhaust Fan & Damper	Yes	Yes	2	4	Automatic start in LOOP.
	Class 1E Electrical Room In-duct heater	Yes	Yes	2	4	Automatic start in LOOP
	CCW Pump Area Air Handling Unit	No	Yes	2	4	

Table 7.4-1 Component Controls for Shutdown
(Sheet 6 of ~~5~~6)

Systems	Components	Normal Shutdown	Safe Shutdown	Train number for Safe Shutdown		Remarks
				Required Number	Actual Number	
HVAC (conti-nued)	Essential Chiller Unit Area Air Handling Unit	No	Yes	2	4	
	EFW Pump Area Air Handling Unit	No	Yes	2	4	
	Essential Chiller Unit	Yes	Yes	2	4	
	Essential Chilled Water Pump & Valves	Yes	Yes	2	4	
	Containment Fan Cooler Unit	Yes	No	-	-	Automatic start in LOOP.
	Reactor Cavity Cooling Fan	Yes	No	-	-	Automatic start in LOOP.
	CRDM Cooling Fans & Unit	Yes	No	-	-	Automatic start in LOOP.
	Non-Class 1E Electrical Room Air Handling Unit & Damper	Yes	No	-	-	Automatic start in LOOP.
	Non-Class 1E Electrical Room Return Air Fan	Yes	No	-	-	
	Non-Class 1E Battery Room Exhaust Fan & Damper	Yes	No	-	-	Automatic start in LOOP.
	Auxiliary Building Air Handling Unit & Damper	Yes	No	-	-	
	MS/FW Piping Area Air Handling Unit & Damper	Yes	No	-	-	
	Non- Essential Chiller Unit	Yes	No	-	-	Automatic start in LOOP.
	Non- Essential Chilled Water Pump & Valves	Yes	No	-	-	Automatic start in LOOP.
	Non-Essential Chiller Condenser Water Pump & Valves	Yes	No	-	-	Automatic start in LOOP.
	Non-Essential Chilled Water System Cooling Tower Fan	Yes	No	-	-	Automatic start in LOOP

Table 7.4-2 Indication for Shutdown

Systems	Instruments	Number of Required Channels	Normal Shutdown	Safe Shutdown	Remarks
RCS	Pressurizer Water Level	2	Yes	Yes	
	Pressurizer Pressure	2	Yes	Yes	
	Reactor Coolant Hot Leg Temperature (Wide Range)	1per Loop	Yes	Yes	
	Reactor Coolant Cold Leg Temperature (Wide Range)	1per Loop	Yes	Yes	
	Reactor Coolant Pressure	1per Loop	Yes	Yes	
CVCS	Boric Acid Tank Water Level	1 per tank	Yes	No	
	RCP Seal Water Return Line Flow	1 per RCP	Yes	No	
	RCP Seal Water Outlet Temperature	1 per RCP	Yes	No	
	Charging Flow	1	Yes	No	
SIS	Safety Injection Pump Discharge Flow	1 per Line	No	Yes	Used to maintain RCS inventory during Safe Shutdown.
	Safety Injection Pump Minimum Flow	1 per Line	No	Yes	
	Safety Injection Pump Discharge Pressure	1 per Line	No	Yes	
	Safety Injection Pump Suction Pressure	1 per Line	No	Yes	
	Accumulator Pressure	1 per Tank	No	Yes	For ACC isolation during Safe Shutdown.
RHRS	CS/RHR Hx Outlet Temperature	1 per Line	Yes	Yes	
	CS/RHR Pump Discharge Flow	1 per Line	Yes	Yes	
	CS/RHR Pump Minimum Flow	1 per Line	Yes	Yes	
	CS/RHR Pump Discharge Pressure	1 per Line	Yes	Yes	
	CS/RHR Pump Suction Pressure	1 per Line	Yes	Yes	
EFWS	EFW Pit Water Level	2 per Pit	No	Yes	
	EFW Flow	1 per Line	No	Yes	
	EFW Pump Discharge Pressure	1 per Line	No	Yes	
CFS	SG Water Level (Wide Range)	1 per SG	Yes	Yes	
MSS	Main Steam Line Pressure	2 per Line	Yes	Yes	
CCWS	CCW Surge Tank Water Level	2 per Tank	Yes	Yes	
	CCW Header Pressure	1 per Line	Yes	Yes	
	CCW Header Flow	1 per Line	Yes	Yes	
	CCW Supply Temperature	1 per Line	Yes	Yes	
ESWS	CCW Hx ESW Flow	1 per Line	Yes	Yes	
	ESW Header Pressure	1 per Line	Yes	Yes	
RWS	RWSP Water Level (Wide Range)	2	No	Yes	
NIS	Source Range Neutron Flux	2	No	Yes	

7.5 Information Systems Important to Safety

7.5.1 System Description

This section describes the I&C systems PSMS and PCMS that provide information to the plant operators for: (1) assessing plant conditions and safety system performance, and making decisions related to plant responses to abnormal events; and (2) preplanned manual operator actions related to accident mitigation. The information systems important to safety also provide the necessary information from which appropriate actions can be taken to mitigate the consequences of AOOs.

This section describes the following information systems important to safety:

- Post accident monitoring (PAM)
- Bypassed and inoperable status indication (BISI)
- Plant annunciators (alarms)
- Safety parameter displays system (SPDS)

Information important to safety, which supports emergency response operations, is available via the emergency response data system (ERDS). Refer to Subsection 7.9.1.7.

The information important to safety is available for display at the following facilities:

- MCR
- RSR
- TSC
- EOF

Controls for credited manual operator actions are available in the MCR.

7.5.1.1 Post-Accident Monitoring

The purpose of displaying PAM parameters is to assist MCR personnel in evaluating the safety status of the plant. PAM parameters are direct measurements or derived variables representative of the safety status of the plant. The primary function of the PAM parameters is to aid the operator in the rapid detection of abnormal operating conditions. As an operator aid, the PAM variables represent a minimum set of plant parameters from which the plant safety status can be assessed.

Safety-related PAM parameters are displayed on the safety VDUs, operational VDUs, and on the LDP. Non safety-related PAM parameters are displayed on operational VDUs. The parameters selected comply with the guidelines of RG 1.97 (Reference 7.5-1). Display of at least two trains of each safety-related parameter is available.

The safety VDUs for each train are isolated from each other and from non-safety systems.

IEEE Std 497-2002 (Reference 7.5-2) provides selecting and categorizing principles for PAM variables. Table 7.5-1 provides a summary of the selection criteria and source documents for each PAM variable type.

Table 7.5-2 provides the US-APWR design attributes for each variable type.

Table 7.5-3 provides a list of PAM variables, their ranges, monitored functions or systems, quality and variable type. To further clarify the US-APWR PAM variable selection basis, Tables 7.5-6 though 7.5-10 show specific PAM variables and their required functions.

The COL applicant is to provide a description of site-specific PAM variables, which are type D variables for monitoring the performance of the UHS and type E variables for monitoring the meteorological parameters.

Instrumentation for monitoring severe accidents is discussed in Subsection 19.2.3.3.7. which summarizes the necessary equipment survivability for achieving and maintaining shutdown of the plant and maintaining containment integrity for severe accidents. A detailed description of the analysis on equipment survivability, including instruments required for severe accident monitoring, is provided in Chapter 15 of PRA Technical Report, MUAP-07030 (Reference 7.5-15)

7.5.1.1.1 Variable Classifications and Signal Processing Design

The following clarifications are provided for the design attributes identified in Tables 7.5-1 and 7.5-2:

(1) Single Failure: The design ensures that at least one measurement channel is available after all ~~credible~~ single failures. Process measurement channels are interfaced to redundant trains of the RPS. Component status signals are interfaced to redundant trains of the SLS. PAM information is then interfaced to redundant safety grade HSI and non-safety HSI for display.

(2) Seismic Qualification: RPS, SLS, and safety grade HSI are seismically qualified, as previously described. PAM measurement channels are generically qualified by the instrument OEM. Specific analysis for the US-APWR demonstrates this qualification bounds the seismic levels for the specific instrument location.

(3) Environmental Qualification: RPS, SLS, and safety grade HSI are environmentally qualified, as previously described. These systems are located in a mild environment; therefore, the qualification duration is not applicable. PAM measurement channels are generically qualified by the instrument OEM. Specific analysis for the US-APWR demonstrates this qualification bounds the environmental conditions for the specific instrument location and required qualification duration. The qualification duration requirements are defined in Section 3.11 for all variable types. For Type C variables monitoring fission product barriers, the qualification duration is a minimum of

physical separation are provided between redundant safety systems and between the safety and non-safety systems.

The system level BISI is provided in the “OK monitor” area on the LDP for inoperable conditions that result in inoperability of any ESF or RT system function at the train level. The BISI is color-coded so that the indication for each function is lighted in yellow color when one train is bypassed, and lighted in red color when two or more trains are bypassed. When the system level BISI are displayed on the LDP, operators can drill down to specific inoperable information in the train level on the operational VDU in the MCR.

With regard to certain items performed at least once per fuel cycle (i.e., up to 24 months while RG 1.47 recommends “per one year”), the system level BISI is automatically initiated by a signal from the PSMS and is not removed by any method until the initiating signal is reset from the PSMS. In addition, to the automatic initiation conditions listed below, the operator can manually initiate the system level BISI from the operational VDU.

- ~~Changing~~ Connecting PSMS controller to the maintenance network~~enable status for the engineering tool~~
- PSMS input bypass to accommodate input calibration and testing
- ESFAS train bypass for testing
- RPS bypass for shunt trip testing
- Component bypass from SLS (to perform component maintenance)
- Bypass or alignment of the components and equipment of the following fluid system in positions that would bypass the safety function (that are tested at least once per 24 months during plant operation)
  - ECCS
  - CS/RHR System
  - EFWS
  - CCWS
  - ESWS
  - HVAC

7.5.1.2.2 Bypassed and Inoperable Status Indication Functions

System level indication for each of the following safety function is displayed.

- RT

components, including audible and visual devices, are redundant to ensure operation is not adversely affected by credible malfunctions. The digital portion of the alarm system integrity is checked by self-diagnosis which does not affect the operation of alarms ~~and digital control system portion of alarms that have self-diagnosis functions~~. Failures in the redundant visual portions of the alarm system are easily identified by operators, since the LDP and alarm VDUs are used routinely by operators for all tasks in the MCR. Failures in the redundant audible portions of the alarm system are easily identified by operators, since distinct alarm sounds normally originate from different locations within the MCR. Alarm signals originate in plant instrumentation or within the controllers of the PCMS and PSMS. These signals are interfaced to the PCMS via the redundant unit bus, described in Section 7.9. The data interface to the PSMS is physically and functionally isolated so as not to affect the safety system in case of failure of the alarm system.

As for all PCMS components, the alarm system is powered by redundant UPSs. The alarm system is designed and tested to a similar environmental, seismic, and EMI/RFI requirement as the PSMS.

The highly reliable design of the alarm system makes it suitable for prompting operator attention to all abnormal plant conditions, including those requiring manual operator actions credited in the plant safety analysis. The alarms for credited manual operator actions are developed through an augmented quality program, which includes software V&V.

7.5.1.4 Safety Parameter Display System

The SPDS provides a display of key plant parameters from which the plant's critical safety function status may be assessed. The primary function of the SPDS is to help operators and emergency response personnel make quick assessments of plant safety status. The SPDS is operated during normal operations as well as during all classes of emergencies. The functions and design of SPDS are included as a part of the overall HSI design. Following is list of SPDS parameters for each critical safety function.

1. Reactivity Control
   - Neutron flux
   - Status of RTBs
   - Control rod position
2. RCS Inventory
   - Pressurizer water level
   - Reactor coolant hot leg temperature (wide range)
   - Reactor coolant cold leg temperature (wide range)
   - Reactor coolant pressure

7.5.1.6.3 Emergency Response Data System

The ERDS, a data transmission system, is designed to send a set of variables from the plant to the NRC operations center. These data may be used for analyses by the NRC headquarters technical support groups and NRC executive team. The ERDS transmits information that will aid NRC in its role of providing advice and support to the nuclear power plant licensee, state and local authorities, and other federal officials.

Communication systems involved with the EOF and using the ERDS are further discussed in Section 7.9.

7.5.2 Design Basis Information

7.5.2.1 Post Accident Monitoring

The PAM design for the US-APWR complies with the requirements of the following codes, standard~~s~~, and RG~~s~~:

- 10 CFR 50, Appendix A: GDC 13 (Reference 7.5-6), 19 (Reference 7.5-7) and 64 (Reference 7.5-8), for specific requirement to provide adequate instrumentation to monitor PA condition(s).
- 10 CFR 50.34(f)(2) “Additional TMI-Related Requirements” (Reference 7.5-9)
  - (xi): regarding direct indication of relief and safety valve position.
  - (xii): regarding auxiliary feedwater system flow indication.
  - (xvii): regarding accident monitoring instrumentation.
  - (xviii): regarding inadequate core cooling instrumentation.
  - (xix): regarding instruments for monitoring plant conditions following core damage.
  - (xx): regarding power for pressurizer level indication.
- RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants” and BTP 7-10 (Reference 7.5-10).
- IEEE Std 497-2002, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations”

IEEE Std 497-2002 contains functional and design requirements for accident monitoring instrumentation for nuclear plant. RG 1.97 endorses IEEE Std 497-2002. For the US-APWR, specific PAM variables comply with the selection criteria described in IEEE Std 497-2002.

7.5.2.2 Bypassed and Inoperable Status Indication

The BISI design for the US-APWR complies with the requirements of the following ~~Codes~~code~~, Standards~~ and RG~~s~~:

- RG 1.47, “Bypassed and inoperable Status Indication for Nuclear Power Plant Safety Systems.”
- 10 CFR 50.34(f)(2)(v), “Additional TMI-Related Requirements” regarding BISI

7.5.2.3 Plant Annunciators

The Plant Annunciators design for the US-APWR complies with the following regulatory guidance:

- Staff Requirements Memorandum (SRM) SECY-93-087, Item II.T, “Control Room Annunciator (Alarm) Reliability. (Reference 7.5-11)

7.5.2.4 Safety Parameter Displays System

The SPDS design for the US-APWR complies with the requirements of the following code~~s, standards,~~ and NUREG~~RGs~~:

- 10 CFR 50.34 (f)(2)(iv), “Additional TMI-Related Requirements” regarding the SPDS console
- NUREG 0737 Supplement 1, “Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability”, with respect to SPDS (Reference 7.5-12)

7.5.2.5 Facilities

The emergency response facility design for the US-APWR complies with the requirements of the following code~~s, standards, and RGs~~:

- 10 CFR 50.34 (f)(2)(xxv), “Additional TMI-Related Requirements” regarding emergency response facilities

7.5.3 Analysis

Detailed compliance with the GDC, IEEE Std 603-1991 (Reference 7.5-13) and IEEE Std 7-4.3.2-2003 (Reference 7.5-14) are described in MUAP-07004 Section 3.0, Appendix A and B.

For most accident conditions, RPS and ESFAS are designed to perform required protective functions automatically without any credit for manual action(s). Manual operator actions are credited for mitigating some accident conditions, as defined in the safety analysis. Manual operator actions are also credited for achieving safe shutdown for normal and post accident conditions.

7.5-5 Safety I&C System Description and Design Process, MUAP-07004-P Rev.3 (Proprietary) and MUAP-07004-NP Rev.3 (Non-Proprietary), September 2009.

7.5-6 Instrumentation and Control, General Design Criteria for Nuclear Power Plant 13, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

7.5-7 Control Room, General Design Criteria for Nuclear Power Plant 19, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

7.5-8 Monitoring Radioactivity Releases, General Design Criteria for Nuclear Power Plant 64, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

7.5-9 Additional TMI-Related Requirements, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50.34(f)(2).

7.5-10 Guidance on Application of RG. 1.97, BTP 7-10 Revision 5, March 2007.

7.5-11 Control Room Annunciator (Alarm) Reliability, SECY-93-087, Item II.T.

7.5-12 Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability, NUREG 0737 Supplement No. 1, January 1983.

7.5-13 IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, IEEE Std 603-1991.

7.5-14 IEEE Standard Design for Digital Computers in Safety Systems of Nuclear Power Generating Stations, IEEE Std 7-4.3.2-2003.

7.5-15 US-APWR Probabilistic Risk Assessment, MUAP-07030 Rev.2 (Proprietary), December 2009.

7.5-16 US-APWR US-APWR Equipment Environmental Qualification Program, MUAP-08015 Rev.0, February 2009.

7.5-17 Task Working Group #4; High-Integrated Control Rooms –Communication Issues(HICRc), Interrim Staff Guidance, DI&C-ISG-04 Revision 1, March 2009.

7.5-18 Guidance for Evaluation of Diversity and Defence-in-Depth in Digital Computer-Based Instrumentation and Control Systems, BTP 7-19 Revision 5, March 2007.

7.5-19 Defense-in-Depth and Diversity, MUAP-07006-P-A Rev.2 (Proprietary) and MUAP-07006-NP-A Rev.2 (Non-Proprietary), September 2009.

7.5-20 Defense in Depth and Diversity Coping Analysis, MUAP-07014-P Rev.~~1~~2 (Proprietary) and MUAP-07014-NP Rev.~~1~~2 (Non-Proprietary), ~~June~~ December 200~~8~~9.

cold shutdown) while below the allowable pressure setpoint. During startup, the valves are manually locked closed (power is removed) to prevent these valves from opening and exposing the RHRS to an over-pressure condition. The valves will not be placed on service again until the subsequent plant cooldown.

The piping connecting the RCS hot leg to RHR pump suction is provided with two MOVs connected in series for each RHR train (8 valves total):

- For RHR train A, the valves are assigned to train A.
- For RHR train B, the valves are assigned to train B.
- For RHR train C, the valves are assigned to train C.
- For RHR train D, the valves are assigned to train D.

Redundant ~~of two~~ (two) valves in series, ensures that over pressurization will not occur even in presence of a single valve failure. The safety interlock prevents valve opening unless the reactor coolant pressure is less than the setpoint for RHR operation.

The signal path for this interlock is from the reactor coolant pressure transmitters to the RPS, and then to the SLS, which controls the MOVs via motor control centers.

CS/RHR pump hot leg isolation valves open permissive interlock does not have independence and diversity in each train, because the current design of RHR system has sufficiently high reliability against overpressurization or possible radioactive release. Moreover the valves cannot be opened inadvertently because power is normally removed as described above.

7.6.1.2 CS/RHR Valve Open Block Interlock

Common CS/RHR pumps are shared between the CSS and RHRS. The CSS and RHRS will not be required at the same time. CSS will be required in the beginning of an AOO or PA to reduce the containment pressure, while the RHR will be employed in the later part of the event to remove decay heat.

- Simultaneous-open block interlock with RHR discharge line containment isolation valve and CS header containment isolation valve;

  Valves are provided for CS/RHR pump discharge for each CS and RHR line. If CS and RHR lines are opened simultaneously, the CS/RHR pump will be loaded beyond its capacity. This could lead to a pump run-out condition, which would damage the CS/RHR pumps. To preclude opening both systems valves simultaneously an interlock is provided to block simultaneous opening of the RHR discharge line containment isolation valve and the CS header containment isolation valve. The interlock functions to prevent opening a valve that is closed. This interlock prevents CS and RHR system from operating simultaneously to prevent a pump run-out situation. The interlocks for these valves are shown in Figures 7.6-2 and 7.6-3. For RHS-MOV-021A, B, C, D, the piping diagrams for these valves are shown in Figure 5.4.7-2 in Chapter 5, and for CSS-MOV-004A, B, C, D in Figure 6.2.2-1 of Chapter 6.

- Simultaneous-open block interlock with CS/RHR pump hot leg isolation valve and CS header containment isolation valve;

  Since the CS/RHR pumps are also used for containment spray, there is a potential for valve misalignment that could lead to pumping RCS inventory through the CS lines. This would result in the inadvertent depletion of RCS inventory. To prevent this condition an interlock is provided to block simultaneous opening of the RCS suction line valves to the CS/RHR pumps and the CS discharge line valves. This interlock prevents CS and RHR systems operating simultaneously, which could lead to inadvertent depletion of RCS inventory. The interlocks for these valves are shown in Figure 7.6-1 and 7.6-3. For RHS-MOV-001A, B, C, D, the piping diagrams for these valves are shown in Figure 5.4.7-2 in Chapter 5, for RHS-MOV-002A, B, C, D in Figure 5.4.7-2 of Chapter 5, and for CSS-MOV-004A, B, C, D in Figure 6.2.2-1 in Chapter 6.

All interlocks discussed above are between valves within the same division:

- For CS/RHR train A, the interlocked valves are assigned to train A.
- For CS/RHR train B, the interlocked valves are assigned to train B.
- For CS/RHR train C, the interlocked valves are assigned to train C.
- For CS/RHR train D, the interlocked valves are assigned to train D.

A single interlock failure may result in valve misalignment within a single division, but this will not adversely affect the other divisions. The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.

The signal path for this interlock is from the valve limit switches to the component control logic for each valve within the SLS.

7.6.1.3 Primary Makeup Water Line Isolation Interlock

The CVCS regulates boron concentration in the RCS by controlling the flow of reactor makeup water from sources that contain primary makeup water and borated water.

Redundant interlocks are provided to close two series isolation valves in the primary makeup water supply flow path. This interlock actuates when the monitored primary makeup water flow exceeds its high setpoint. This interlock blocks primary makeup water supply flow, preventing over dilution of the RCS. The interlocks for primary makeup water line isolation valves are shown in Figure 7.6-4. For CVS-FCV-128, 129 the piping diagrams for these valves are shown in (Figure 9.3.4-1 (Sheet 4 of 7)) in Chapter 9.

The two-train redundancy of this design provides over dilution protection even in the presence of a single failure. The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.

The signal path for this interlock is from local flow transmitters to the RPS, and then to the SLS, which controls each isolation valve.

7.6.1.4 Accumulator Discharge Valve Open Interlock

Each of the four RCS loops is provided with a separate accumulator. Each ECCS accumulator discharge line connecting to the RCS cold leg is provided with a motor operated isolation valve. Normally the isolation valve is open; therefore, the accumulator system is normally available for its designed function.

The accumulator discharge valve can be closed manually. However, an interlock is provided to open this valve when the reactor coolant pressure is above the P-11 setpoint. The interlocks for these valves are shown in Figure 7.6-5. The safety related interlocks precludes multiple valve misalignment due to spurious commands from Operational VDU.

This interlock may be manually bypassed for test and maintenance to close the accumulator discharge valve by two deliberate operator actions. If this valve is closed and not selected to "~~Pull~~ Lock", then the ECCS actuation signal will automatically open the valve and make the accumulator system available. The "~~Pull~~ Lock" function is described in Topical Report MUAP-07007 (Reference 7.6-1) Section 4.5.3.a.

The accumulator system can be bypassed for test and maintenance by manually closing its discharge valve and selecting it to "~~Pull~~ Lock". In the "~~Pull~~ Lock" mode, the accumulator discharge valves will not automatically open, therefore the affected accumulator will be un-available for its designed ESF function. During this condition, the inoperable status of the accumulator is alarmed in the MCR and indicated continuously on the BISI system displays.

The signal path for this interlock is from the pressurizer pressure transmitters to the RPS, and then to the SLS, which controls these MOVs via motor control centers.

7.6.1.5 CCW Supply and Return Header Tie Line Isolation Interlock

The CCW system consists of two independent subsystems. Each subsystem consists of two 50% trains. One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four 50% trains. There are cross-connections between trains A and B, and between trains C and D. Each subsystem supplies a non-essential safety class loop and a non-safety loop. There are two series motor-operated isolation valves for each supply and return tie line between separate trains. These isolation valves ensure each mechanical safety train is isolated from any potential passive failure in the non-safety portion or another mechanical safety train of the CCWS.

The two series isolation valves in each CCW header are automatically closed during the following conditions:

- ECCS actuation combined with LOOP
- CS actuation
- Low CCW surge tank water level

For NCS-MOV-007A, B and NCS-MOV-020A, B the piping diagrams for these valves are shown in Figure 9.2.2-1 (Sheet 1 of 9) in Chapter 9, and for NCS-MOV-007C, D and NCS-MOV-020C, D in Figure 9.2.2-1 (Sheet 2 of 9) of Chapter 9.

The interlocks for these valves are shown in Figure 7.6-6. These interlocks ensure the independence of each safety mechanical train of the CCWS thereby providing CCW coolant to ESF systems required for mitigating conditions of the event. The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.

These interlocks may be manually bypassed for reopening the valves to restore RCP seal and spent fuel pit heat exchanger cooling, if required. The bypass can be selected from the safety VDU. To select the bypass from the Operational VDU, the Bypass Permissive for the respective train must be enabled.

Two series valves assigned to different trains ensures isolation even in the presence of a single failure.

The signal path for the ECCS and CS interlocks is from the ESFAS to the SLS that controls the isolation valves through motor control centers. The signal path for the surge tank interlock is from local level transmitters to the RPS to the SLS for control of these same valves.

7.6.1.6 RCP Thermal Barrier HX CCW Return Line Isolation Interlock

Each CCW subsystem supplies cooling water to the RCP thermal barrier heat exchanger. Two motor-operated valves and flow meters are located at the CCW outlet line of the RCP thermal barrier heat exchanger.

These valves close automatically upon a high flow rate signal at the outlet of this line in the event of in-leakage from the RCS through the thermal barrier heat exchanger, and prevent this in-leakage from further contaminating the CCWS.

The interlocks for these valves are shown in Figure 7.6-7. The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.

These interlocks ensure isolation of in-leakage from the RCS through the thermal barrier heat exchanger.

The signal path for these interlocks is from the local flow transmitters to the RPS, and then SLS, which controls these MOVs via motor control centers.

7.6.1.7 Low-pressure ~~l~~Letdown ~~l~~Line ~~i~~Isolation Interlock~~ valve~~

A single normally closed air-operated valve is placed in each of the two low pressure letdown lines connected to two of the four RHR trains. During the normal plant cool down operation, one of these valves is open to divert a portion of the RCS flow to the CVCS for the purpose of purification and RCS inventory control.

Additionally at mid-loop operation during plant shutdown, these valves are automatically closed and the CVCS is isolated from the RHRS after receiving the RCS loop low-level signal to prevent loss of RCS inventory.

The interlocks for these valves are shown in Figure 7.6-8. For RHS-AOV-024B and 024C the piping diagrams for these valves are shown in Figure 5.4.7-2 in Chapter 5. The safety related interlocks preclude multiple valve misalignment due to spurious commands from Operational VDUs.

The signal path for these interlocks is from the local pressure transmitters to the RPS, and then SLS, which controls these MOVs via motor control centers.

7.6.2 Design Basis Information

The interlock systems important to safety comply with the following codes and standards:

1. 10 CFR 50.55a(a)(1), “Quality Standards.”
2. 10 CFR 50.55a(h), “Protection and Safety Systems,”
3. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, “Quality Standards and Records.”
4. GDC 2, “Design Bases for Protection Against Natural Phenomena.”
5. GDC 4, “Environmental and Dynamic Effects Design Bases.”
6. GDC 10 “Reactor Design”
7. GDC 13, “Instrumentation and Control.”
8. GDC 15 “Reactor Coolant System Design”
9. GDC 16 “Containment Design”
10. GDC 19, “Control Room.”
11. GDC 20 “Protection System Functions”
12. GDC 21 “Protection Systems Reliability and Testability”
13. GDC 22 “Protection System Independence”

7.7 Control Systems Not Required for Safety

The function of the US-APWR control systems not required for safety is to establish and maintain the plant operating conditions within prescribed limits. These control systems improve plant safety by minimizing the frequency of protection responses required and relief the operator from routine tasks.

The control functions not required for safety are implemented by the PCMS. The PCMS regulates conditions in the plant automatically in response to changing plant conditions and changes in plant load demand. These operating conditions include the following:

- Step load changes of plus or minus 10% while operating in the range of 15 to 100% of full power.
- Ramp load changes of plus or minus 5% per minute while operating in the range of 15 to 100% of full power (subject to core power distribution limits)
- Full load rejection from 100% power

These capabilities are accomplished without a reactor trip. Full load rejection is an event in which the main generator is cut off from the transmission system ~~either~~ by a tripping of the main transformer breaker or the switchgear breaker without causing a turbine trip~~, or by a trip of the main generator~~. In a load rejection scenario, the turbine governor valves are immediately fully closed, and the turbine bypass valves are opened fully, dumping the excess steam in the condenser. Reactor power is decreased by the automatic insertion of the control rods.

The AOOs defined in the plant safety analysis that must be considered in the control function design are listed in Table 7.7-1. To ensure the PCMS failures do not cause the concurrent AOOs that have not been considered in the plant safety analysis, control functions are distributed to separate the PCMS controller groups as shown in Table 7.7-2. The following sections describe the control functions and the features of those functions that ensure credible control system failures are bounded by the plant safety analysis.

7.7.1 Description

The following sections describe US-APWR control functions not required for safety that can affect the performance of critical safety functions.

7.7.1.1 Reactor Control System

The reactor control system section in the PCMS provides the following automatic functions to respond to the load changes described above.

7.7.1.1.1 Rod Control

The rod control function controls the reactor coolant average temperature (T_{avg}) by sending control signals to the CRDM control system to adjust control rod bank positions,

- Over power ΔT
- Over temperature ΔT
- Turbine inlet pressure
- Control rod bank D position

To generate these interlocks the PCMS receives neutron flux signals from the RPS via the unit bus. Two high intermediate range neutron flux signals are used in 1-out-of-2 logic, since there are only two intermediate range neutron flux detectors for low power conditions. Four power range neutron flux signals are used for normal power conditions. The power range neutron flux logic is 1-out-of-4 to ensure rod withdrawal is stopped to prevent abnormal flux distribution in local areas of the core, which may occur as a result from other rod control malfunctions (i.e., rod withdrawal after compensating from a dropped rod event). The CRDM control system performs this function by receiving this signal (via the path above). This ensures that one detector can prevent this abnormal flux distribution quickly, since the other detectors may realize this event later than the excore detector affected by the local flux distribution.

The over power ΔT and over temperature ΔT input signals are interfaced from the RPS to the PCMS via the unit bus. Within the PCMS, signals from each of the four RPS trains, corresponding to each of the four RPS loops, are processed through the SSA within the PCMS before being used to generate these interlocks.

The turbine inlet pressure input signal is interfaced from the RPS to the PCMS via the unit bus. Within the PCMS, signals from each of the four RPS trains are processed through the SSA within the PCMS before being used to generate these interlocks.

All interlocks block automatic or manual control rod withdrawal. These interlocks are provided from the reactor control section of the PCMS to the CRDM control system.

The over power ΔT and over temperature ΔT interlock also initiates a turbine runback. These interlocks are provided from the reactor control system in the PCMS to the BOP control system.

The over power and over temperature interlocks~~,~~ which block rod withdrawal, are generated from a separate controller group from the rod control function discussed above~~,~~ which generates rod control withdrawal demands. This improves the potential for stopping inadvertent rod withdrawals that may be generated due to failures in the CRDM control group.

Plant operators are alerted by alarms and indications to conditions of control system malfunctions and/or abnormal operating conditions.

7.7.1.1.3 Control Rod Bank Insertion Limit Alarms

The PCMS generates control rod bank insertion limit alarms to alert the operator of excessive rod insertion, refer to Figure 7.2-2 sheet 16. The alarms prompt the operator control mode. In the manual mode the operator can fix the charging flow control valve position.

7.7.1.1.8 Low Pressurizer Water Level Interlock

An interlock is provided to prevent excessive low pressurizer water level conditions that could result from excessive letdown or inadequate charging initiated by either a control system malfunction or operator violation of operating procedures, refer to Figure 7.2-2 sheet 20.

To generate this interlock the PCMS receives pressurizer water level signals from the RPS and processes these signals through SSA, as discussed above.

This interlock automatically closes letdown line isolation valves #1 and #2. The interlock also de-energizes the backup heaters to prevent damage during low pressurizer water level conditions where they may become uncovered.

This interlock is provided from the reactor control system in the PCMS to backup heater switchgear and the control solenoid on letdown line isolation valve #1 and #2.

One of the low pressurizer water level interlocks is generated from a separate controller group from the pressurizer water level controls which generates charging flow demands. This improves the potential for preventing excessive low pressurizer water level conditions that may be generated due to failures in the PCMS pressurizer water level control group. Plant operators are alerted by alarms and indications to conditions of control system malfunctions and/or abnormal operating conditions.

7.7.1.1.9 Steam Generator Water Level Control

Water level in the shell side of the SGs is maintained by the SG water level control function at a pre-determined setpoint, refer to Figure 7.2-2 sheet 21. During normal plant transients, the SG water level is controlled to prevent an undesirable reactor trip.

Three modes of the SG water level control function are provided:

- During normal power operation, three-element feedwater control regulates flow of MFW flow into the SGs via the MFW line with the MFW regulation valve by continuously comparing the SG water level signal, the fixed level reference, the MFW flow signal, and the steam flow signal.

- During low-power operation, two-~~three-~~element feedwater control regulates the flow of MFW into the SGs, bypassing the MFW regulation valve with the MFW bypass regulation valve by continuously comparing ~~the MFW flow signal,~~ the SG water level signal, the fixed level reference, and the reactor coolant ΔT signal.~~ A separate low range feedwater flow measurement is used in the low-power SG water level control mode.~~

- During hot standby operation, single element feedwater control regulates the flow of MFW into the SGs, also bypassing the MFW regulation valve and MFW error signal reduces in magnitude, indicating that the T_{avg} is being reduced toward the reference no-load value, the turbine bypass valves are modulated. This regulates the rate of decay heat removal and establishes the equilibrium hot shutdown condition.

7.7.1.1.12 Turbine Bypass Interlock

Low-low T_{avg} turbine bypass block:

The turbine bypass control functions are prevented by the ~~this~~ interlock from the PSMS, which controls redundant non-Class 1E ~~safety-related~~ permissive solenoids on each turbine bypass valve from SLS trains A and D. Excessive T_{avg} cooldown is blocked by this function as described in Subsection 7.3.1. ~~5.~~12.

Loss of load interlock:

Actuation of turbine bypass on small load perturbations is prevented by an independent load rejection sensing circuit. This circuit senses the rate of decrease in the turbine load as detected by turbine inlet pressure. It unblocks the turbine bypass valves only when the rate of load rejection exceeds the preset value corresponding to a 10% step load decrease or a sustained ramp load decrease of greater than 5% per minute. The unblocking of the turbine bypass valves is latched to enable the load rejection operating mode. This latch is manually reset by the operator after plant stabilization using the loss of load reset switch.

Condenser not available interlock:

This non-safety interlock is implemented in the PSMS to simplify the interface with the non-Class 1E~~safety-related~~ turbine bypass valve permissive solenoids used for the T_{avg} interlock described above. Turbine bypass valve permissive solenoids are controlled by the PSMS to achieve high reliability of block turbine bypass function as described in Subsection 7.3.1.12.

These interlocks improve the potential for preventing inadvertent turbine bypass conditions that may be generated due to failures in the PCMS turbine bypass control group.

7.7.1.2 Nuclear Instrumentation System

The RT signals derived from the nuclear instrumentation are described in Section 7.2. The control functions which use these same safety-related nuclear instrumentation signals are described in the sections above.

In addition, the reactor control system in the PCMS provides the following non-safety nuclear instrumentation monitoring functions:

- Indicated nuclear power.
- Indicated axial flux difference.

- Upper radial power tilt alarm on signal levels from the upper half of the power range neutron flux detector.
- Lower radial power tilt alarm on signal levels from the lower half of the power range neutron flux detector.
- Axial flux difference alarm deviation exceeds the limits.

In the MCR, provisions are made to continuously indicate and record nuclear power, axial flux imbalance and signal levels for each power range neutron flux detector.

7.7.1.3 Control Rod Drive Mechanism Control System

The CRDM control system in the PCMS~~,~~ adjusts the position of the control rod banks in the reactor core. Each control rod bank is divided into two or more groups to obtain smaller incremental reactivity changes per step. The control rod groups within the same bank are moved such that the relative position of the groups does not differ by more than one-step. Each control rod in a group is paralleled so that rods of the same group move simultaneously.

Power to the CRDMs is supplied by two motor-generator sets operating from two separate 480 V, three-phase busses. Each generator is the synchronous type, and is driven by an induction motor. The ac power is distributed to the CRDM control system power cabinet through the RTBs.

The CRDM control system consists of a logic cabinet and power cabinet, both located in close proximity to the CRDM motor generator sets. The PCMS controller group of the CRDM control system is located within the logic cabinet. The controller group controls solid-state CRDM power supplies that are located in the power cabinet.

Manual control is provided from the OC to move individual control rods or entire control rod banks in or out of the core. Control rod speed for manual control is fixed at approximately 48 steps per minute. The length of a step is described in Subsection 3.9.4.

The CRDM control system provides control for control rod shutdown banks and control rod control banks.

There are four control rod shutdown banks. The control rod shutdown banks are manually withdrawn to the full-out position prior to the reactor becoming critical. The control rod shutdown banks are always in the fully withdrawn position until the reactor is tripped or shutdown.

There are four control rod control banks, which are positioned to control reactor power after the control rod shutdown banks are fully withdrawn. The control rod control banks may be manually controlled or automatically controlled by the rod control function, refer to Subsection 7.7.1.1.1. In the manual mode, control banks may be moved individually or in a pre-determined overlap sequence. In the automatic mode, the control banks are withdrawn or inserted in the same predetermined overlapped sequence.

7.7.2.1 Safety Classification

The PCMS is a non safety-related system. The plant accident analysis of Chapter 15 does not rely on the operability of any PCMS control functions to assure safety. Safe shutdown can be achieved without reliance on any PCMS control functions.

7.7.2.2 Effects of Control System Operation on Accidents

For the transient response of the plant systems for AOOs and PAs, the safety analysis takes no credit for normal PCMS control actions that would lessen the affects of the event (e.g., reduction of feedwater by the SG water level control system during a SG tube rupture event). In addition, the safety analysis assumes normal control actions, that would aggravate the affects of the event and are not blocked by safety functions, will occur (e.g., increase of charging flow by the pressurizer water level control system during a SG tube rupture event).

7.7.2.3 Effects of Control System Failures

The Chapter 15 analysis of AOOs bounds all ~~credible~~ single random failures within the PCMS. This includes single failures that result in:

- A fail as-is, fail de-energized or spurious actuation of a single PCMS hardware component (e.g., input module, or output module).
- A fail-as-is or fail de-energized condition of an entire PCMS control group; the control function to control group assignment are shown in Table 7.7-2.
- Spurious actuation of a single or multiple control functions (e.g., reactivity control, pressurizer control, or SG water level control) within a control group, resulting from a single software block failure.
- A spurious single command from an operational VDU.
- Stuck or dropped control rod
- Stuck control rod bank or overlap sequence error
- Spurious actuation of a normal rod motion command (spurious motion of any single bank)
- Spurious motion of multiple control banks in the predetermined overlap sequence.

The Chapter 15 analysis of AOOs, credits the affects of interlocks in PCMS control groups not affected by the failure, which limit the affects of a failed PCMS control group or control function.

The following types of failures are not considered credible, since they require a series of specific successive failures in multiple software blocks:

- Multiple spurious commands from an operational VDU. Since multiple spurious commands from an operational VDU are not credible, they are not considered in the analysis of bounding AOOs. However, multiple spurious commands from an operational VDU are analyzed for their effect on the safety functions, in MUAP-07004 Appendix D.

- Spurious actuation of multiple control functions in the same control group that do not rely on the same software block.

- Spurious actuation of un-programmed control actions (e.g., out of sequence motion for multiple control banks).

- Spurious motion of a single control rod.

MUAP-07004 (Reference 7.7-1) Section 5.1.8 describes the basis for the PCMS failures assumed in safety analysis.

7.7.2.4 Effects of Control System Failures Caused by Accidents

The PCMS controllers are in mild environment locations, which are not impacted by plant accidents. In addition, most PCMS inputs come from safety-related sensors, which are qualified for accident environments. To accommodate random PCMS failures and PCMS failures that may be caused by accident conditions, the Chapter 15 safety analysis assumes the worst case PCMS single failure, which would aggravate the accident condition and is not blocked by safety functions.

7.7.2.5 Environmental Control System

Environmental control systems that are credited in the safety analysis are controlled by the PSMS, not the PCMS. Environmental control systems controlled by the PCMS, such as non-essential area HVAC, heat tracing, and/or forced air-cooling or heating, are considered in the failure analyses described above, refer to Subsections 7.7.2.3 and 7.7.2.4.

7.7.2.6 Use of Digital Systems

The PCMS and PSMS utilize the same basic software. In addition, the PCMS application software is developed using a structured process similar to that applied to development of the PSMS application software. This process includes an augmented quality program, including software V&V, for the following functions:

- Safety functions controlled by operational VDUs

- SPDS

- Alarms for credited manual operator actions

- SSA

Therefore, the potential for control system failures that could challenge safety systems or impact plant safety functions has been minimized.

7.7.2.7 Independence

The PCMS is physically, electrically, and functionally independent of PSMS, refer to Subsection 7.1.3.4 and 7.1.3.5 for related details.

7.7.2.8 Defense-In-Depth and Diversity

PCMS and PSMS utilize the MELTAC digital platform, which is described in ~~Topical Report~~ MUAP-07005 (Reference7.7-2). Maximum utilization of a common digital platform throughout a nuclear plant reduces maintenance, training, and changes due to obsolescence, thereby minimizing the potential for human error.

The potential for CCF in these systems is minimized by the following:

- Simplicity of the basic design.
- Maturity of the MELTAC platform.
- Design process including the elevated quality programs applied to both systems.
- Significant functional diversity within the numerous computers that compose these systems.

Regardless of this very low potential for CCF, the DAS is provided to accommodate beyond design basis CCFs that could adversely affect the PSMS and PCMS concurrent with an AOO or PA, refer to Section 7.8 for DAS details.

7.7.2.9 Potential for Inadvertent Actuation

The PCMS design limits the potential for inadvertent actuation and challenges to the PSMS as follows:

- The PCMS processes multiple redundant sensor signals through the SSA. The SSA ensures the PCMS does not take erroneous control actions based on a single instrument channel failure or single RPS train failure.
- The PCMS includes interlocks that limit erroneous control actions, refer to applicable Subsections in 7.7.1.1 above. These interlocks are assigned to control groups that are distinct from other control groups, which can initiate erroneous control actions.
- The PCMS control functions are distributed to multiple control groups such that erroneous actions resulting from a control group failure are bounded by the plant safety analysis.

- Operational VDUs generate control commands based on two distinct operator actions, in accordance with ISG-04 Position 3.1.5.

7.7.2.10 Control of Access

Security-Related Information – Withheld Under 10 CFR 2.390

~~Cyber security control of the PCMS is described in Subsection 7.9.2.6.~~

7.7.3 Analysis

The Chapter 15 analysis for AOOs and PAs does not take credit for operability of the PCMS for accident/event mitigation or achieving and maintaining safe shutdown. In addition, PCMS failures are bounded by the Chapter 15 analysis. Refer to Subsections 7.7.2.2 through 7.7.2.4.

The plant transient analysis demonstrates the control systems are capable of safely controlling the plant, without the need for manual intervention and without violating plant protection or component limits, for the following:

- 10% step load change while operating in the range of 15% to 100% of full power without RT or turbine bypass system actuation.
- Ramp load changes at 5% power per minute while operating in the range of 15% to 100% of full power without RT or turbine bypass system actuation (subject to core power distribution limits).
- Full-load rejection without RT.

The control system permits maneuvering the plant through the above transients without actuation of the following:

- Main Steam safety valves
- Safety depressurization valves

In addition, these valves are not actuated during a normal plant trip.

7.7.4 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

7.7.5 References

7.7-1 Safety I&C System Description and Design Process, MUAP-07004-P Rev.3 (Proprietary) and MUAP-07004-NP Rev.3 (Non-Proprietary), September 2009.

7.7-2 Safety System Digital Platform -MELTAC-, MUAP-07005-P Rev.4 (Proprietary) and MUAP-07005-NP Rev.4 (Non-Proprietary), September 2009.

7.7-3 Highly Integrated Control Rooms – Digital Communication Systems, DI&C-ISG-04 Revision 1, March 2009.

Table 7.7-2 Controller Group Control System Distribution in the Reactor Control System

Group	Control System	Postulated Event Due to a Single Failure in the Corresponding Controller Group*1			
		Full open of any of the MFW regulation valves	Full open of any one of the main steam relief valves or turbine bypass valve	Two banks of control rod withdrawal according to bank overlap sequence	Full open of the charging flow control valve or pressurizer spray valve
Group 1	A-SG Feedwater Control	X			
	A-Main Steam Relief Valve Control		X*2		
Group 2	B-SG Feedwater Control	X			
	B-Main Steam Relief Valve Control		X*2		
	Pressurizer Pressure Control				X*2
Group 3	C-SG Feedwater Control	X			
	C-Main Steam Relief Valve Control		X*2		
	Pressurizer Water Level Control				X*2
	Control Rod Insertion Monitoring			X	
Group 4	D-SG Feedwater Control	X			
	D-Main Steam Relief Valve Control		X*2		
	~~Control Rod Insertion Monitoring~~			~~X~~	
Group 5	Turbine Bypass Control		X*2		
	Reactor Makeup Control			X	
Group 6	Control Rod Control			X*2	

Note:
1. This table describes that one controller failure does not cause credible failures in other controller groups.
2. An interlock is provided (for this control system) in a separate controller group, to limit the effect of the single controller failure.

7.8 Diverse Instrumentation and Control Systems

The DAS is the non-safety diverse instrumentation and control system for US-APWR. The DAS provides monitoring, control and actuation of safety and non-safety systems required to cope with abnormal plant conditions concurrent with a CCF that disables all functions of the PSMS and PCMS. The DAS includes an automatic actuation function, HSI functions located at the diverse HSI panel (DHP), and interfaces with the PSMS and PCMS. The design basis and detailed system description for the DAS are described in Topical Report MUAP-07006 (Reference 7.8-1). Table 7.8-7 shows the supplemental information to Topical Report MUAP-07006-P-A, which is necessary to be clarified. The Defense in Depth and Diversity Coping Analysis, Technical Report MUAP-07014 (Reference 7.8-2), demonstrates the ability to maintain all critical safety functions and achieve hot standby using the DAS.

The DAS design consists of conventional equipment that is totally diverse and independent from the MELTAC platform of the PSMS and PCMS, so that a beyond design basis CCF in these digital systems will not impair the DAS functions. In addition, the DAS includes internal redundancy to prevent spurious actuation of automatic and manual functions due to a single component failure. The DAS is also designed to prevent spurious actuations due to postulated earthquakes and postulated fires. The DAS interfaces with the safety process inputs and outputs of the SLS are isolated within these safety systems. In addition, hardwired Class 1E logic within the SLS (not affected by a CCF) ensures that control commands originating in the DAS or SLS, which correspond to the desired safety function, always have priority. Therefore, there is no adverse interaction of the DAS with safety functions and no erroneous signals resulting from CCF in the SLS that can prevent the safety function. For a figure of the DAS system architecture, refer to Figure 6.0-1 of Topical Report, MUAP-07006.

Within the DAS, manual actuation is provided for systems to maintain all critical safety functions (Refer to Table 7.8-1). For conditions where there is insufficient time for manual operator action, the DAS provides automatic actuation of required plant safety functions needed for accident mitigation. Key parameter indications, diverse audible and visual alarms, and provisions for manual controls are located in a dedicated independent DHP located in the MCR. Conventional hardwired logic hardware and relays for automatic actuation are installed~~located~~ in two diverse automatic actuation cabinets (DAACs), each located in a separate room. Each DAAC is powered by a separate non-Class 1E UPS. During plant on-line operation, the system can be tested manually without causing component actuation that would disturb plant operations.

7.8.1 System Description

The DAS consists of manual HSI functions, which include~~ diverse leak detection, and~~ automatic actuation functions. These functions are located in the DHP and the DAAC, respectively. In addition, the DAS consists of interfacing connections with the PSMS and CRDM motor-generator sets. The DAS receives inputs from qualified analog isolators located in the RPS or directly from plant components. The DAS provides outputs which interface to the SLS power interface modules via qualified isolators located in the SLS or directly to plant components.

The manual actuation switches listed above are sufficient to take all manual actions credited in Technical Report MUAP-07014, which demonstrates the ability to maintain all critical safety functions and achieve hot standby. Hot standby can be maintained for an extended period-of-time by direct operation of local power distribution and switching devices that are not affected by the CCF in the PSMS.

7.8.1.1.2 Alarms

When the DAS system level actuation signals are generated for (1) reactor trip, turbine trip, and MFW isolation, or for (2) EFW actuation are generated, a summary alarm for these functions is also actuated on the DHP. The diverse audible alarm is activated to notify the operators. The first out alarm panel, on the DHP, indicates the specific input parameter that has caused the system level actuation.

Failure information about the DAS, such as power supply failure, or module de-energization or removal, is alarmed as a "DAS failure summary alarm" on the Alarm VDU in the MCR. The configuration of the DAS alarms is described in Topical Report MUAP-07006 Section 6.2.2.1. High main steam radiation (N16) and high-high steam generator water level are alarmed and indicated on DHP. Technical Report MUAP-07014 provides the specific information of the alarm credited for D3 coping analysis.

7.8.1.1.3 Indicators

The analog indicators provided on the DHP are identified in Table 7.8-2. These indicators are sufficient to support all manual control actions credited in Technical Report MUAP-07014, which demonstrates the ability to maintain all critical safety functions, and achieve and maintain hot standby.

7.8.1.2 Diverse Automatic Actuation Cabinet

Each DAAC provides for automatic actuation of critical systems, which are required to be actuated within first 10 minutes of an event (refer to Table 7.8-3 for system actuation times). The defense in depth and diversity coping analysis provides justification for manual operator actions credited after 10 minutes.

Safety sensors selected by the plant design for the DAS input are interfaced from within the PSMS or PCMS input modules. These input modules utilize analog distribution modules and isolation modules that connect the input signals to the DAS prior to any digital processing. Therefore, a software CCF within the PSMS or PCMS does not affect the DAS automation function or the display of plant parameters on the DHP. The MELTAC input module design of the PSMS or PCMS is described in ~~Topical Report~~ MUAP-07005 (Reference 7.8-4) Section 4.0.

The DAS has two analog logic subsystems, one each located in one of the two DAACs.

Within each DAAC, input signals are compared to their setpoint values and if the monitored value is greater than or less than its setpoint, a partial trip/actuation signal is generated. RT signals and/or ESF actuation signals are generated from each DAAC through voting logic of its input signals. The voting logic (2-out-of-4) for each specific

To support the single failure criterion for all PSMS functions, there are four SG water level signals (one per each train A, B, C, and D) on each SG. However, for the DAS, which does not need to meet the single failure criterion, only one water level signal is required from each SG.

The reactor trip is actuated by tripping the non-safety CRDM motor-generator set. This actuation leads to de-energizing the power for the CRDM by a means that is diverse from the RTB to release the control rods for gravity insertion into the reactor core. Diversity from the PSMS is maintained from sensor-inputs to final actuators.

The Turbine Trip is actuated by opening the solenoid valves for turbine trip. Diversity from the RT function in the PSMS is maintained from sensor-input up to the power interface module.

The MFW isolation is actuated by closing the MFW regulation valve. Diversity from the feedwater isolation function in the PSMS is maintained from sensor input up to the power interface module.

These DAS actuation functions are automatically blocked when all the following conditions are established:

- Status ~~status~~ signals are received indicating that the minimum combination of the RTBs have actuated for the RT function. This is referred to as the P-4 interlock. The logic for the P-4 interlock is the same as in the PSMS, as shown in Figure 7.8-2. The P-4 interlock is processed independently in each DAAC. Signals from all RTBs are interfaced from the PSMS, prior to any software processing, to each DAAC, as shown in Figure 7.8-1.

- The turbine emergency trip oil pressure trip signal is generated when oil pressure channels exceed the trip setpoint.

7.8.1.2.2 Emergency Feedwater Actuation

EFW is automatically actuated on a low SG water level signal. 2-out-of-4 voting logic is utilized for the low SG water level signals from each SG.

The interface and configuration of the SG water level signals is as described above.

Diversity from the EFW actuation function in the PSMS is maintained from sensor input up to the power interface module. This automatic DAS EFW function is automatically blocked when status signals are received indicating that the PSMS EFW function has actuated correctly. Correct actuation is indicated when 2-out-of-4 status signals are received from limit switch contacts on the steam inlet valves to the turbine driven EFW pumps and from auxiliary contacts on the motor starters controlling the motor driven EFW pumps, as shown in Figure 7.8-3. The EFW pump status signals are interfaced from the PSMS, prior to any software processing, to each DAAC, as shown in Figure 7.8-1.

- Minimize the potential for CCF
- Cope with CCF for AOOs
- ~~Cope with software CCF for PA~~
- ~~Minimize the extent of software CCF~~
- ~~Minimize the effects of software CCF~~
- ~~Minimize the potential for adverse interaction~~

Das is implementd to mitigate the adverse effects/impacts from digital I&C both hardware and software common cause failure (CCF). It is not to minimize the potential or extent of software CCF.

A detailed description of each principle is provided in Topical Report MUAP-07006 Section 5.0.

7.8.2.9 Fire Protection

Fire protection for the DAS is described in MUAP-07004 Sections 5.2.3 and 6.5.8.

7.8.3 Analysis

7.8.3.1 Anticipated Transient without Scram

In accordance with 10 CFR 50.62 (Reference 7.8-6), the DAS is diverse from the RT system to initiate turbine trip and EFW actuation. Although not required by 10 CFR 50.62 for all reactors, MHI's defense in depth and diversity approach also includes a diverse RT function for ATWS mitigation.

A detailed discussion of conformance of to 10 CFR 50.62 is provided in Topical Report MUAP-07006 Appendix B.

7.8.3.2 Adequacy of Manual Controls and Displays

Technical Report MUAP-07014 defines the alarms, indicators and controls required on the DHP for the operator to take manual actions credited for mitigating each AOO and PA included in Chapter 15, place the nuclear plant in a hot standby condition, and monitor and control the following critical safety functions:

- Reactivity control
- Core heat removal
- Reactor coolant inventory control
- Containment integrity

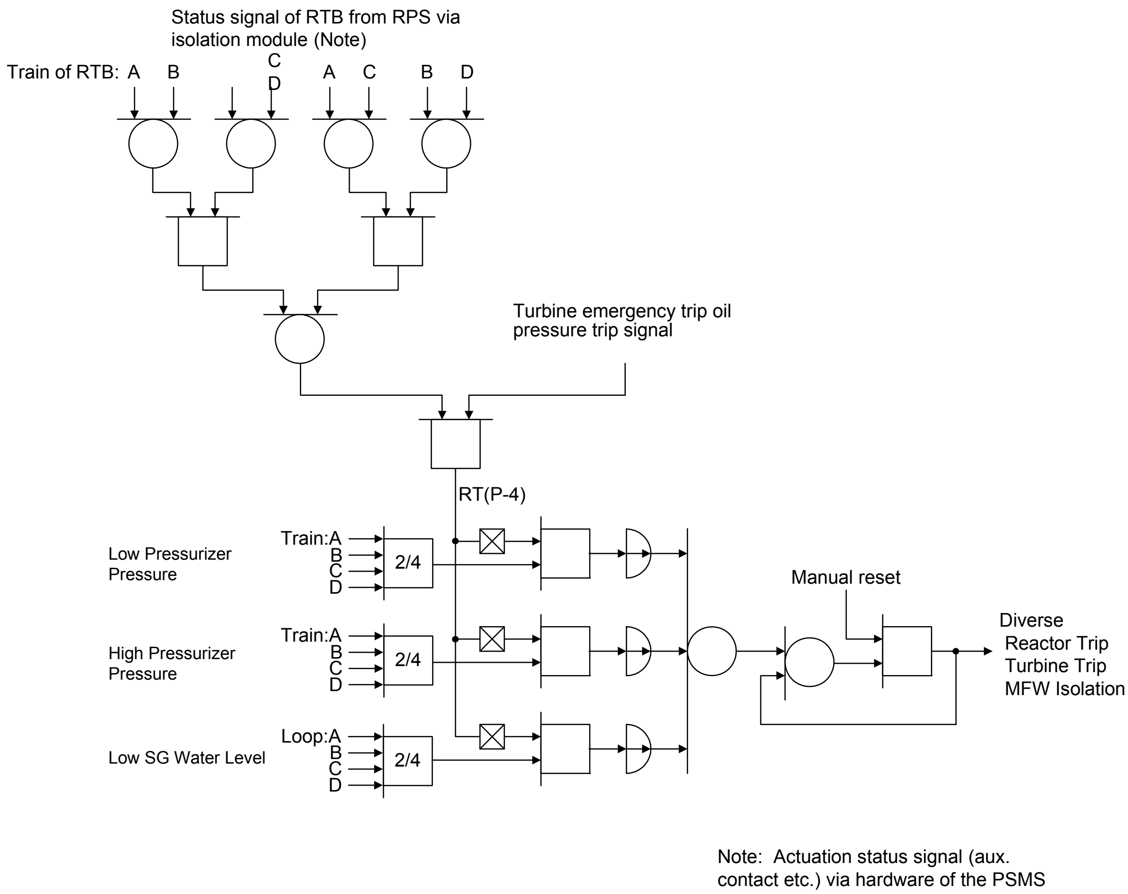


Figure 7.8-2 The Prevention Diagram of Reactor Trip, Turbine Trip and MFW Isolation in DAS

7.9 Data Communication Systems

The DCS consists of the plant-wide unit bus, safety bus for each PSMS train, maintenance network for each PSMS train and the PCMS (five maintenance networks total), data links for point-to-point communication, and I/O bus for each controller. Figure 7.1-1 shows the involvement of each part of the DCS in the overall I&C architecture, with the exception of the maintenance network, which is shown in ~~Topical Report~~ MUAP-07005 (Reference 7.9-1) Section 4.3.4. The DCS interfaces to the station bus, which is an information technology network (i.e., not I&C), as described in Subsection 7.9.1.6. The DCS interfaces to the HSIS and the unit bus are shown in Figure 7.9-1 and described in Table 7.9-1.

Although the DCS is a distributed and highly interconnected system, there is communication independence to prevent electrical and communication processing faults in one division (safety or non-safety) from adversely affecting the performance of safety functions in other divisions. To prevent electrical faults from transferring between divisions and between different plant fire areas for the MCR, RSR and I&C rooms, qualified fiber-optic isolators are used. Communication faults are prevented through a data integrity verification process that is described in ~~Topical Report~~ MUAP-07005 Section 4.3.2.

7.9.1 System Description

7.9.1.1 Control Network (Safety Bus and Unit Bus)

The control network technology is utilized for the plant-wide unit bus and the safety bus for each PSMS division.

The control network is a redundant bi-directional multi-node network that has automatic node detection and redirection features. The network provides the functionality to maintain communication even if a controller is powered down or if there is an unresponsive controller.

The Control Network is a fiber optic network, which interfaces to the control network interface module of each controller via qualified electrical to optical (E/O) converter. There is a separate E/O converter and control network interface module for each control network (i.e., unit bus and safety bus).

Details on the control network are described in Topical Report MUAP-07005 Section 4.3.2.

7.9.1.1.1 Safety Bus

The controllers of the SLS, ESFAS, and RPS, and the safety VDU processor communicate via the safety bus within each PSMS division. The signals transmitted on this network are described in Sections 7.2 through 7.6. The interconnections of these systems to the safety bus are shown in Figures 7.2-1 and 7.3-1.

There is one safety bus for each train. Each safety bus is used only within the same train.

7.9.1.1.2 Unit Bus

The unit bus provides non-safety data communication between all I&C systems. The main signals transmitted through the unit bus are:

- Manual operation signals transmitted from the operational VDUs in the MCR and RSR to the PSMS and PCMS. Signals to the PSMS are blocked by automated safety signals and logic in the PSMS, which ensures priority of all safety functions. All safety components controlled by the PSMS have automated safety signals and priority logic.

- Signals transmitted from the PCMS to PSMS for interlocks and automatic control of safety components during normal operation. These signals are blocked by automatic safety signals logic in the PSMS, which ensures priority of all safety functions. All safety components controlled by the PSMS have automated safety signals and priority logic.

- Process and alarm signals transmitted from the PSMS and PCMS to the LDP and VDUs in all operating locations, MCR, RSR, and TSC and to the computer systems such as process recording computer system, alarm processor system, etc.

- Shared sensor signals, such as pressurizer pressure, and shared calculated signals, such as T_{avg}, are transmitted from each PSMS division to the PCMS.

Signals interfaced between the PSMS and PCMS use qualified E/O isolators that are part of the safety system. In addition, communication and functional isolation are provided, within the PSMS, for signals sent from the PCMS to PSMS, such as process control signals and signals from operational VDUs. These signals are interfaced via redundant communication subsystems within the PSMS, referred to as the COM, which provide the communication interface between the unit bus and all PSMS controllers for non-safety control signals that originate in the PCMS.

Further details on communication independence are discussed in MUAP-07004 (Reference 7.9-2) Appendix B.5.6.

7.9.1.2 Safety VDU Communication

The safety VDU has two interfaces with the safety VDU processor:

- Conventional red/green/blue video signals are interfaced through a frame memory unit module within the safety VDU processor.

- The safety VDU touch panel is interfaced to the safety VDU processor through a touch panel interface module, which provides a conventional point-to-point data link.

Safety VDU processors are located in the Class 1E I&C room. There are separate safety VDU processors for the safety VDUs in the MCR and the safety VDUs in the RSR. Each safety VDU processor interfaces to the safety bus by a qualified E/O isolator. This isolator ensures that electrical faults originating in the MCR or the RSR, which may adversely affect the respective safety VDU Processors, cannot propagate to the safety bus. Therefore, these faults cannot adversely affect safety bus communications or other controllers/processors on the safety bus. (The location of the safety VDU processors is different from the location described in ~~Topical Report~~ MUAP-07005.)

Control commands from the safety VDU processors are interfaced to other PSMS processors in the same train via the Safety Bus and the COM. Within the COM, the safety VDU commands are combined with control commands from Operational VDUs. The priority logic in the COM ensures safety VDU commands always have priority over corresponding Operational VDU commands. In addition, this logic allows all Operational VDU commands to be blocked when the Safety VDU “Disconnect” command is selected, as shown in MUAP-07004 Figure 5.1-3.

7.9.1.3 Data Links

Data link communication is used to transmit signals between the controllers in different divisions and within controllers of the same division, as follows:

- Between RPS controllers in all trains
- Between RPS and ESFAS controllers in all trains
- Between the reactor control system and CRDM control system controllers
- Between incore instrumentation system and the unit management computer

Data links use fiber optic interfaces to provide electrical isolation between divisions. Separate E/O conversion devices are used at the receiving end and sending end for each data link interface.

The data link interface provides unidirectional broadcast only communication with no data communication handshaking. Communication independence is assured by two port memory and specific attributes of the basic software within the controllers. These design features ensure that communication with external divisions cannot disrupt the deterministic processing of control functions, including the safety functions of the PSMS. ~~Topical Report~~ MUAP-07005 Section 4.3.3 provides a detailed description of the data links including communication independence.

Data links are interfaced to the controller via the bus master module. The bus master module has four ports, which can be configured for either sending or receiving, as follows:

- Between RPS controllers in all trains - Each RPS controller includes one bus master module configured to broadcast its data to the RPS controllers in the three other trains (one of four ports utilized). This same bus master module receives broadcast data from the RPS controllers in the three other trains (three of four ports utilized).

- Between RPS and ESFAS controllers in all trains - Each RPS controller includes one bus master module configured to send its data to the two ESFAS controllers in its own train and the two ESFAS controllers in the three other trains (one of four ports utilized). Each ESFAS controller includes two bus master modules to receive the broadcast data from the eight RPS controllers in all four trains. One bus master module receives data from RPS controllers A - Group 1, A - Group 2, B - Group 1, and B - Group 2 (four of four ports utilized), the second bus master module receives data from RPS controllers C - Group 1, C - Group 2, D - Group 1, and D - Group 2 (four of four ports utilized).

The failure of a bus master module and E/O conversion device is considered in the FMEA.

7.9.1.4 I/O Bus

The I/O bus provides a bi-directional interface between a controller and its I/O modules. The I/O bus is interfaced via the bus master module in the controller and the repeater module within each I/O chassis. For single non-redundant controller configurations, the I/O bus is not redundant. For redundant controller configurations, the I/O bus is redundant. Various redundancy configurations are utilized as described in ~~Topical Report~~ MUAP-07005 Section 4.1.1.1.

I/O can be located in close proximity to the controller or in locations remote from the controller. Remote I/O is utilized for both PCMS and PSMS applications.

7.9.1.5 Maintenance Network

The maintenance network is a non-safety system that allows for monitoring the status of the PSMS and PCMS equipment failure indications and diagnostics, updating setpoints and constants, and the installation of new application software. PSMS controllers are normally not connected with the maintenance network. PSMS controllers that are temporarily connected to the maintenance network are declared inoperable and the affected inoperable functions of that controller are managed by Technical Specifications. Access control for the maintenance network is described in Technical Report MUAP-07004 Section 6.4.1. ~~–~~There is communication independence for the maintenance networks for each division. However, since all maintenance networks are non-safety, no electrical independence is required and there are locations in the plant where all maintenance networks are in close physical proximity. The following description is applicable to the maintenance network for any one division.

The major components of the maintenance network are the switching hub and the engineering tool. The maintenance network interfaces to the system management module of each controller via qualified E/O converters.

The engineering tool is a dedicated non-safety personal computer, which runs on the Microsoft Windows operating system (OS). It contains MELTAC software, which allows it to interact with the controller via the maintenance network. The engineering tool is continuously connected to the maintenance network~~, which is continuously connected to the controllers of its division. (This continuous connection is different from the temporary connection of the engineering tool described in Topical Report MUAP-07005.) This continuous connection is assured by the use of the qualified isolators, which provide physical and electrical independence. Additionally, communication independence is assured by two port memory and specific attributes of the basic software within the controllers. These design features ensure that communication with the engineering tool cannot disrupt the deterministic processing of control functions or the safety functions of the PSMS. Topical Report MUAP-07005 Section 4.1.4.2 provides a detailed description of communication independence for the engineering tool~~.

When a MELTAC controller is temporarily connected to the maintenance network, The engineering tool can be~~is normally~~ used for monitoring~~ purposes only. However, it can also be used to change application setpoints and constants and update controller software~~MELTAC controller performance, self-testing diagnostics and functional logic execution. The PSMS application setpoints, constants and application software are changeable only by removing the CPU module that contains the memory devices from the MELTAC controller and placing it in a dedicated reprogramming chassis. When the dedicated reprogramming chassis is connected to the engineering tool, either directly or via the maintenance network, the engineering tool is used to down load changes. The software installation procedure verifies the authenticity and integrity of the application software through a software installation procedure, described in ~~Topical Report~~ MUAP-07005 Section 6.1. The PSMS basic software is changeable only by removing and replacing the memory device that contains the software.

~~In order to update constants and setpoints or allow software installation, the controller is locally selected to write-enable, using a conventional hardware write permission switch. The details of the hardware based switch enable function are described in Topical Report MUAP-07005 Section 4.3.4.2.~~ There are several physical security requirements to allow controllers to be connected to the maintenance network or to allow software changes ~~this~~ to occur, including key locked cabinet doors, door open alarms in the MCR, and alarms in the MCR when a controller is connected to the maintenance network or powered down to allow CPU module removal~~for the write-enable mode~~. In addition, technical specifications ensure that functions affected by powering down a PSMS controller~~s~~ or connecting it to the maintenance network are declared inoperable in accordance with Technical Specifications~~by plant operators prior to enabling the write-enable mode~~.

There are multiple engineering tools connected to the maintenance network via the switching hub. An engineering tool is located in each of the I&C rooms. In addition, an engineering tool for each division is centrally located in the plant maintenance facility.

7.9.1.6 Station Bus

The station bus provides information to plant and corporate personnel and to the EOF and ERDS. The station bus receives information from the DCS via the unit management computer. The unit management computer provides a firewalled interface, which allows only outbound communication. There are no other connections from external sources to the DCS.

7.9.1.7 External Network Interface

The only interface from the PCMS and PSMS to external networks is via the firewall within the unit management computer. The unit management computer provides an outbound only interface to the plant Station Bus to allow communication to EOF computers, the NRC (via ERDS), corporate information systems and plant personnel computers.

7.9.2 Design Basis Information

7.9.2.1 Quality of Components and Modules

The PSMS includes the safety bus, data links, I/O bus, and safety VDU communications. The MELTAC platform is applied for all safety DCS components and follows the MELCO QA program. The quality of PSMS components and modules and the quality of the PSMS design process is controlled by a program that meets the requirements of ASME NQA-1-1994 (Reference 7.9-3). Conformance to ASME NQA-1-1994 is described further in Chapter 17.

The PCMS includes the unit bus, data links, I/O Bus, and the PCMS computers. The PCMS data communications uses the same hardware as the PSMS. The PCMS has a similar quality program to the PSMS, without the same level of documentation.

7.9.2.2 Software Quality

The safety related portions of the DCS are part of the PSMS. The non-safety related portions of the DCS are part of the PCMS. All portions of the DCS consist of MELTAC basic software, which handles the communication protocol and self-diagnostics, and application software, which handles the actual data being transmitted.

MHI applies its MELCO's safety system digital platform MELTAC to PSMS and PCMS systems of US-APWR. Details of the software quality program for the MELTAC basic software are discussed in ~~Topical Report~~ MUAP-07005 Section 6.0. A summary of the software quality program for the PSMS application software is discussed in ~~Topical Report~~ MUAP-07004 Section 6.0. A description of the application software quality program is provided in the Software Program Manual for US-APWR Technical Report MUAP-07017 (Reference 7.9-4).

The Software Program Manual Technical Report MUAP-07017, describes the processes, which ensure the reliability and design quality of the PSMS application software throughout its entire software lifecycle. MUAP-07017 also provides the software program plans based on the guidance of BTP 7-14. By following this SPM, the PSMS application software achieves high functionality and high quality including data communication systems as follows.

- Application software for the PSMS achieves a quality level expected for nuclear plant safety functions.
- Application software provides the required safety functions.
- The processes and procedures described in MUAP-07017 are based on established technical and document control requirements, practices, rules and industrial standards.

7.9.2.3 Performance Requirements

DCS in digital I&C system of the US-APWR meets the performance of required functions. The performance of the digital I&C system including DCS conforms to the guideline of BTP 7-21(Reference 7.9-15). Technical Report MUAP-09021 (Reference 7.9-16) provides the response time of safety I&C system. The report demonstrates that the safety I&C system meets the response time requirement from safety analysis. The simplified block diagrams of the RT and ESF functions propagation paths and response time of each path in the safety I&C system are provided. The conformance of BTP 7-21 and how the safety I&C system meets the performance requirements are also addressed in MUAP-09021.

7.9.2.3.1 System Deterministic Timing

All DCS communication protocols allow calculation of a deterministic data communication response time. The time calculation includes the number of nodes on the network, data traffic, network topology, node processing cycle time, and network throughput. The methods used for real-time performance calculations are described in ~~Topical Report~~ MUAP-07005 Section 4.4.

7.9.2.3.2 Real-Time Performance

Real-time performance is determined by performing response time analysis for all safety functions. For each safety function an analysis has been performed which demonstrates the actual system response time is less than the response time required by the plant safety analyses. Refer to ~~Topical Report~~ MUAP-07004 Section 6.5.2 for the related details. Response times for the RPS and ESFAS functions are listed in Tables 7.2-3 and 7.3-4 respectively.

7.9.2.3.3 Time Delays within the DCS

Data propagation delays due to data communication in the DCS are incorporated into response time analysis. Response time calculations, which encompass the controller and all components connected to the DCS, include these data propagation delays. DCS response time calculations are validated through sample tests, during system integration testing, refer to ~~Topical Report~~ MUAP-07004 Section 6.5.3.

7.9.2.3.4 Data Rates and Bandwidth

The data rates and bandwidths for the sections of the DCS are listed in ~~Topical Report~~ MUAP-07005 as follows:

- Control network: Table 4.3-2.
- Data links: Section 4.3.3.
- Maintenance network: Section 4.1.4.2.
- I/O bus: Appendix A.3.
- Safety VDU communication: Appendix A.11 and A.12.

7.9.2.3.5 Interfaces with other DCS

The only interface from the DCS to external networks is via the firewall within the unit management computer. The unit management computer provides an outbound only interface to the plant station bus to allow communication to the EOF computers, the NRC (via ERDS), corporate information systems, and plant personnel computers.

7.9.2.3.6 Test Results

MELTAC platform testing demonstrates that the DCS meets all generic qualification requirements, refer to ~~Topical Report~~ MUAP-7005 Section 5.0. Qualification analysis demonstrates that the generic qualification testing bounds all US-APWR conditions.

The PCMS and PSMS factory test phase demonstrates that the DCS meets all US-APWR application performance requirements, refer to ~~Topical Report~~ MUAP-07004 Section 6.1.

7.9.2.3.7 Communication Protocols

All communication protocols selected for the DCS are able to support all demands from interfacing systems. Refer to ~~Topical Report~~ MUAP-07005 Section 4.0 and Appendix A for further details on the specific communication protocols used in each network of the DCS including capabilities, bandwidth, and data rates.

7.9.2.4 Potential Hazards and Single Failures

The self-diagnostic features described in ~~Topical Report~~ MUAP-07004 Section 4.3, detect DCS errors or failures. The MELTAC controller has separate self-diagnostic features for each of the DCS related modules as described in ~~Topical Report~~ MUAP-07005 Section 4.1.5 and Section 4.3. All DCS errors and failures are analyzed in the FMEA, which demonstrates that there are no single failures that can result in loss of the safety function. The FMEA identifies errors or failures that can result in failures or inadvertent actuation of single divisions, which are bounded by the plant safety analysis.

In addition, the safety controllers within the PSMS include electrical and communication isolation to ensure that the deterministic processing of the safety functions can not be affected due to failures or communication errors from the unit bus or maintenance network. Table 7.2-8 and Table 7.3-7 which shows the FMEA for reactor trip and ESF actuation in the PSMS include failure mode and effects of the DCSs.

7.9.2.5 Control of Access

Security-Related Information – Withheld Under 10 CFR 2.390

7.9.2.6 Cyber Security

The use of computer systems for various functions at nuclear power plants including digital I&C systems increases the potential for threats from cyber intrusions.

~~The PSMS and PCMS (including the unit management computer), and all computers connected to the station bus are controlled within the plant's cyber security program which meets the requirements of Nuclear Energy Institute (NEI) 04-04 (Reference 7.9-11).~~

~~Refer to Technical Report MUAP-08003 "US-APWR Cyber Security Program" (Reference 7.9-12) for details.~~

~~The cyber security program ensures the plant's critical systems are protected against cyber threats and includes but is not limited to:~~

- ~~Cyber security program including defensive strategy based on the industrial guidance, NEI 04-04.~~
- ~~Cyber security applicable to the US-APWR digital safety system based on regulatory guidance, RG 1.152 (Reference 7.9-13) and BTP 7-14 (Reference 7.9-14).~~
- ~~Protecting all critical systems from cyber threats during design, operation and maintenance phase of their life cycle.~~
- ~~Graded approach with varying levels of security based on the significance of the system function.~~
- ~~Identification of the requirements for each level of cyber security.~~
- ~~Approach to mitigate risk for each level of cyber security.~~
- ~~Good industry practices for the development phases of the software life cycle for non-safety systems to ensure systems delivered to plant are free of cyber threats.~~
- ~~Plant equipment as well as programmatic aspects such as administration, procedures and personnel training.~~
- ~~Programmatic requirements for ongoing periodic assessment of the effectiveness of the program, including ongoing assessment of cyber threats.~~

The COL applicant is to provide a description of cyber security provisions.

7.9.2.7 Independence

The DCS ensures electrical independence between PSMS divisions and between the PSMS and PCMS. In addition, electrical independence is maintained within the PSMS and PCMS, where the communication interfaces cross fire areas of the MCR and RSR.

Each PSMS and PCMS controller/processor protects itself against DCS errors or failures that could disrupt its internal application functions, thereby ensuring communications independence. For more detailed discussion on the methods used to ensure independence between digital systems in different safety trains and between safety and non-safety systems refer to Subsections 7.1.3.4 and 7.1.3.5, and ~~Topical Report~~ MUAP-07004 Appendix A.5.6 and Appendix B.5.6.

All PSMS DCS cables, with the exception of its maintenance networks, are routed in accordance with IEEE Std 384-1992 (Reference 7.9-5) to ensure physical independence of each division. PSMS maintenance network cables, which are non-safety, are routed with other non-safety cables, including PCMS DCS cables.

7.9.2.8 Fail Safe Failure Modes

In general, controllers take no automatic fail-safe actions in response to failures in the unit bus, safety bus, data links, or I/O Bus. This means that inputs to control algorithms are considered to remain in the state prior to the DCS failure, and outputs from controllers remain in their state prior to the DCS failure. All DCS failures are alarmed so that operators can take appropriate manual actions. These actions may include setting signals to trip or maintenance bypass status, and declaring appropriate Technical Specification inoperable status. Where failures affect only one of two redundant buses, actions may be limited to only initiating work orders for maintenance repairs.

The RPS controllers take fail-safe actions in response to failures in multiple data links, between the RPS trains, that result in loss of data to the 2-out-of-4 voting logic within each train. On the first failure (or bypassed data link) the voting logic becomes 2-out-of-3. On the second failure (or bypassed data link) the voting logic becomes 1-out-of-2. Failure (or bypass) of a third link will generate a trip based on this 1-out-of-2 logic. In addition, outputs from the RPS controllers to the RTBs will fail in a state that initiates opening of the RTBs if there is a failure of the I/O bus. This design satisfies the fail-safe requirements of 10 CFR 50 Appendix A, GDC 23 (Reference 7.9-6).

Alarms are provided for the DCS failures. Unique alarms are provided for DCS failures that affect inputs to the RPS voting logic from multiple divisions. For example, unique alarms are provided if an RPS controller detects two or more data link failures or a single data link failure when one of its own process parameters is already in a maintenance bypass condition.

7.9.2.9 System Testing and Surveillances

The MELTAC controller has separate self-diagnostic features for each of the DCS related modules; refer to ~~Topical Report~~ MUAP-07005 Section 4.1.5 and Section 4.3. There are no periodic manual surveillance tests required for DCS functions.

7.9.2.10 Bypass and Inoperable Status Indications

There are no manual bypasses for any functions of the DCS. DCS failures are alarmed on the operational and alarm VDUs.

7.9.2.11 EMI/RFI Susceptibility

The PSMS DCS is qualified to the EMI/RFI testing requirements of RG 1.180 (Reference 7.9-7), refer to ~~Topical Report~~ MUAP-07005 Section 5.3.

The PCMS DCS uses the same hardware and software components as the PSMS DCS.

7.9.2.12 Defense-In-Depth and Diversity

There is no credit for continued the DCS operability in the defense in depth and diversity coping analysis (i.e., the DCS is assumed to fail due to CCF). The DCS is not used by the conventional analog and hardwired DAS. A discussion on defense in depth and diversity is provided in Topical Report MUAP-07006 (Reference 7.9-8).

7.9.2.13 Seismic Hazards

All safety-related DCS components and hardware are Class 1E qualified and are in an appropriately qualified structure. Where non-safety portions of the DCS interface with the safety portions, qualified isolators are used which preserve the seismic qualifications of the safety-related portions. Refer to ~~Topical Report~~ MUAP-07005 Section 4.1 and 5.2 for the related details.

The operational VDUs and unit bus are also tested to demonstrate operability after an SSE. In addition, the testing demonstrates that there are no erroneous signals generated that can adversely affect the PSMS or PCMS systems.

7.9.3 Analysis

Detailed compliance to the GDC, IEEE Std 603-1991 (Reference 7.9-9) and IEEE Std 7-4.3.2-2003 (Reference 7.9-10) are described in ~~Topical Report~~ MUAP-07004 Section 3.0, Appendix A and B.

The FMEA demonstrates that failures in the DCS do not adversely affect the safety function of the PSMS or cause erroneous safety function actuation, refer to ~~Topical Report~~ MUAP-07005 Section 7.4.

7.9.4 Combined License Information

~~No additional information is required to be provided by a COL applicant in connection with this section.~~

COL 7.9(1) *~~Deleted~~The COL applicant is to provide a description of cyber security provisions*

7.9.5 References

7.9-1 Safety System Digital Platform -MELTAC-, MUAP-07005-P Rev.4 (Proprietary) and MUAP-07005-NP Rev.4 (Non-Proprietary), September 2009.

7.9-2 Safety I&C System Description and Design Process, MUAP-07004-P Rev.3 (Proprietary) and MUAP-07004-NP Rev.3 (Non-Proprietary), September 2009.

7.9-3 Quality Assurance Program Requirements for Nuclear Facilities, ASME NQA-1-1994.

7.9-4 Software Program Manual, MUAP-07017 Rev.0, December 2007.

7.9-5 Criteria for Independence of Class 1E Equipment and Circuits, IEEE Std 384-1992.

7.9-6 Protection System Failure Modes, General Design Criteria for Nuclear Power Plant 23, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

7.9-7 Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems, Regulatory Guide 1.180 Revision 1, October 2003.

7.9-8 Defense-in-Depth and Diversity, MUAP-07006-P-A Rev.2 (Proprietary) and MUAP-07006-NP-A Rev.2 (Non-Proprietary), September 2009.

7.9-9 IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, IEEE Std 603-1991.

7.9-10 IEEE Standard Design for Digital Computers in Safety Systems of Nuclear Power Generating Stations, IEEE Std 7-4.3.2-2003.

7.9-11 ~~Cyber Security Program for Power Reactors, NEI 04-04 Revision 1, November 2005.~~ Intentionally Blanked

7.9-12 ~~US-APWR Cyber Security Program, MUAP-08003-P Rev.0 (Proprietary) and MUAP-08003-NP Rev.0 (Non-Proprietary), August 2008.~~ Intentionally Blanked

7.9-13 Criteria for Digital Computers in Safety Systems of Nuclear Power Plants, Regulatory Guide 1.152 Revision 2, January 2006.

7.9-14 Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems, BTP 7-14 Revision 5, March 2007.

7.9-15 Guidance on Digital Computer Real-Time Performance, BTP 7-21 Revision 5, March 2007.

7.9-16 Response Time of Safety I&C System, MUAP-08021-P Rev.0 (Proprietary) and MUAP-09021-NP Rev.0 (Non-Proprietary), October 2009.

Chapter 8

US-APWR DCD Chapter 8 Rev.2, Tracking Report Rev.7 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.1-1	8.1.1	Delete “The plant’s high voltage switchyard is sitespecific and is not part of the reference plant design.”
8.1-5	8.1.5.1 1st paragraph	Editorial: Added “,” Deleted “are site-specific.” Replaced “ T ” with “ t ” Deleted “at the plant site are also site-specific. These site-specific items”
8.1-5	8.1.5.1 4th paragraph	Editorial: Added “, P1 and P2” Added “, P1 and P2” Added “, P1 and P2”
8.1-10	8.1.5.3.5	Editorial: Deleted “offsite and” Added “The switchyard and transmission system design conforms to the standards discussed in Section 8.2.”
8.1-11	8.1.5.3.5	Editorial: Deleted “IEEE Std 518-1982, “IEEE Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources”” Deleted “IEEE Std 524a-1993, “IEEE Guide to Grounding During the Installation of Overhead Transmission Line Conductors”” Deleted “IEEE Std 525-2007, “IEEE Guide for the Design and Installation of Cable Systems in Substations””

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.1-13	8.1.5.3.5	Editorial: Deleted "IEEE Std 980-1994, "IEEE Guide for Containment and Control of Oil Spills in Substations"" Deleted "IEEE Std 998-1996, "IEEE Guide for Direct Lightning Stroke Shielding of Substations"" Deleted "IEEE Std 1159.3-2003, "Recommended Practice for the Transfer of Power Quality Data"" Deleted "IEEE Std 1247-2005, "IEEE Standard for Interrupter Switches for Alternating Current, Rated Above 1,000 V""
8.1-14	8.1.5.3.5	Editorial: Deleted "IEEE Std C37.013-1997, "IEEE Standard for AC High-Voltage Generator Circuit Breakers Rated on a Symmetrical Current Basis"" Deleted "IEEE Std C37.20.3-2001, "Standard for Metal-Enclosed Interrupter Switchgear""

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.1-15	8.1.5.3.5	Editorial: Deleted “IEEE Std C37.29-1981, “IEEE Standard for Low-Voltage AC Power Circuit Protectors Used in Enclosures”” Deleted “IEEE Std C37.30-1997, “IEEE Standard Requirements for High Voltage Switches"” Deleted “IEEE C37.32-2002, “High-Voltage Switches, Bus Supports, and Accessories –Schedules of Preferred Ratings, Construction Guidelines and Specifications”” Deleted “IEEE Std C37.46-2000, “American National Standard for High Voltage Expulsion and Current-Limiting Type Power Class Fuses and Fuse Disconnecting Switches”” Deleted “IEEE Std C37.47-2000, “American National Standard for High Voltage Current-Limiting Type Distribution Class Fuses and Fuse Disconnecting Switches”” Deleted “IEEE Std C37.61-1973, “IEEE Standard Guide for the Application, Operation, and Maintenance of Automatic Circuit Reclosers”” Deleted “IEEE Std C37.73-1998, “Standard Requirements for Pad-Mounted Fused Switchgear””
8.1-16	8.1.5.3.5	Editorial: Deleted “IEEE Std C37.122-1993, “IEEE Standard for Gas-Insulated Substations”” Deleted “IEEE Std C37.123-1996, “IEEE Guide to Specifications for Gas-Insulated, Electric Power Substation Equipment””
8.1-18	Table 8.1-1 (Sheet 2 of 7) DCD Section / Subsection, Remark	Editorial: Deleted “A”. Added “Not applicable ”

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.2-1	8.2.1.2 3rd paragraph	Editorial: Deleted “air cooled”. Added “cooled by forced air with water cooling.” Deleted “air cooled”.
8.2-2	8.2.1.2 5th paragraph	Editorial: Deleted “and” Added “P1 and P2”
8.2-3	8.2.1.2 11th paragraph	Editorial: Replaced “27.3” with “28”
8.2-4	8.2.1.2 12th paragraph	Editorial: Deleted “and non-Class 1E P buses”
8.2-8	8.2.2.1	Editorial: Replaced “is designed and tested in accordance with” with “conforms to the requirement of the”
8.3-6	8.3.1.1.2.2	Editorial: Added “Pressurizer heater Back-up groups are supplied from the Class 1E power systems based on 10CFR50.34(f) (2) (xiii).”
8.3-2	8.3.1.1.1 6thparagraph	Editorial: Deleted “and” Replaced “and RAT to UAT for non-Class 1E MV buses P1 and P2” with “P1 and P2”

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-25	8.3.1.1.6	Other Replaced "In case of failure of the UPS unit or if the UPS unit is out on maintenance, buses A, B, C and D are switched to the 50kVA, 480V/120V ac bypass transformer associated with the same train. Switching circuits are provided with contactors for transfer between the UPS unit power and the transformer power. When the input power of switching circuit from UPS unit is lost, undervoltage relay actuates. Following the undervoltage signal, a contactor of UPS unit side is opened, and then a contactor of transformer side is closed with time delay, automatically. Administrative controls ensure that no more than one vital ac bus is powered from the bypass transformer at any time during routine preventive maintenance of the associated UPS unit. The transfer from the transformer back to the UPS is performed manually." with "Each UPS and bypass transformer is connected to the 120V ac distribution panel. Each bypass transformer is connected to the 120V ac bus via switching circuit when the inverter is not in service. Normally the 120V ac bus is fed from the inverter. When the inverter fails or is out on maintenance, the 120V ac bus is transferred to the bypass transformer by static switch or manually through synchronizing, without interruption of power to the loads. The static bypass switch has the capability of automatically retransferring the load back to the inverter after its output has returned to normal."
8.3-26	8.3.1.1.7	Deleted "all"
8.3-28	8.3.1.1.8	Deleted "two" Replaced "electrical rooms" with "area local."
8.3-38	8.3.1.2.2	Editorial: Replaced "never" with "not" Added "except briefly during recovery from SBO." Added "and recovery from SBO."
8.3-48	8.3.2.2	Editorial: Added ", 33, 34, 35, 38, 41, 44"

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-56	8.3.2.3.5	Editorial: Replaced “8.3.1.3.5” with “8.3.1.3.6”
8.3-63	Table 8.3.1-1 item4	Editorial: Added “(Cooling air is cooled by water)” Replaced “27.3” with “28”
8.3-63	Table 8.3.1-1 item5	Editorial: Added “(Cooling air is cooled by water)” Replaced “27.3” with “28”
8.3-66	Table 8.3.1-3 (Sheet 1 of 3)	Editorial: Added “Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.”
8.3-67	Table 8.3.1-3 (Sheet 2 of 3)	Editorial: Added “Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.”
8.3-68	Table 8.3.1-3 (Sheet 3 of 3)	Editorial: Added “Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.” Replaced “41250” with “14250”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-83	Table 8.3.1-9 (Sheet 4 of 7)	Name of Motor Control Center are changed. Replaced “N11” with “N31” Replaced “N21” with “N41” Replaced “N31” with “N51” Replaced “N41” with “N61” Bldg/F and Elevation are changed for following equipment. - N21-UPS Unit Replaced “2F” with “1F” Replaced “34'-0"” with “3’-7"” - N21-Non-Class 1E AC120V Switch Board Replaced “2F” with “1F” Replaced “34'-0"” with “3’-7"” - N22-UPS Unit Replaced “3’-7"” with “34'-0"” - N22-Non-Class 1E AC120V Switch Board Replaced “3’-7"” with “34'-0"”

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-85	Table 8.3.1-9 (Sheet 6 of 7)	Bldg/F and Elevation are changed for following equipment. - N4-Non-Class 1E 480V Load Center Replaced “1F” with “2F” Replaced “3’-7"” with “34'-0"” - P12-Non-Class 1E Motor Control Center Replaced “1F” with “2F” Replaced “34'-0"” with “3’-7"” Replaced “Electrical Room” with “Turbine Building Local” - P22-Non-Class 1E Motor Control Center Replaced “Electrical Room” with “Turbine Building Local” - N1-Non-Class 1E Motor Control Center Replaced “2F” with “3F” Replaced “34'-0"” with “61'-0"” Replaced “Electrical Room” with “Turbine Building Local” - N2-Non-Class 1E Motor Control Center Replaced “1F” with “3F” Replaced “3’-7"” with “61'-0"” Replaced “Electrical Room” with “Turbine Building Local” - N32-Non-Class 1E Motor Control Center Replaced “2F” with “3F” Replaced “34'-0" with “61'-0"” Replaced “Electrical Room” with “Turbine Building Local” - N42-Non-Class 1E Motor Control Center Replaced “2F” with “1F” Replaced “34'-0" with “3’-7"” Replaced “Electrical Room” with “Turbine Building Local”

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-85	Table 8.3.1-9 (Sheet 6 of 7)	- N52-Non-Class 1E Motor Control Center Replaced “1F” with “3F” Replaced “3’-7"” with “61'-0"” Replaced “Electrical Room” with “Turbine Building Local” - N62-Non-Class 1E Motor Control Center Replaced “Electrical Room” with “Turbine Building Local”
8.3-87	Table 8.3.1-10	Quantity of Emergency feed water pump actuation valve is changed from “1” to “2”. Current is changed from “25 A” to “50 A” Total Current is changed “308 A” to “333 A”
8.3-89	Table 8.3.2-1 (Sheet 1 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-90	Table 8.3.2-1 (Sheet 2 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-91	Table 8.3.2-1 (Sheet 3 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-92	Table 8.3.2-1 (Sheet 4 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-94	Table 8.3.2-2 (Sheet 1 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-95	Table 8.3.2-2 (Sheet 2 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-96	Table 8.3.2-2 (Sheet 3 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-98	Table 8.3.2-2 (Sheet 4 of 4)	Editorial: Added “Note: The DC loads are preliminary and typical, and are subject to change during detailed design.”
8.3-99	Table 8.3.2-3 (Sheet 1 of 2)	RAI No 670-4773 QUESTION 09.04.05-14 Added “2 strings of”
8.3-103	Figure 8.3.1-1 (Sheet 2 of 7)	Editorial: Rated output of B-EMERGENCY FEED WATER PUMP is changed from “450kW” to “590kW”.
8.3-104	Figure 8.3.1-1 (Sheet 3 of 7)	Editorial: Rated output of C-EMERGENCY FEED WATER PUMP is changed from “450kW” to “590kW”.
8.3-107	Figure 8.3.1-1 (Sheet 6 of 7)	Editorial: Deleted “D-CONTAINMENT FAN COOLER”
8.3-110	Figure 8.3.1-2 (Sheet 2 of 24)	Editorial: The power sources are as shown below: 6.9kV P1-Bus: UAT3 and RAT3 6.9kV P2-Bus: UAT4 and RAT4 13.8kV N1-Bus: UAT1 and RAT1 13.8kV N2-Bus: UAT2 and RAT2
8.3-133	Figure 8.3.1-3 (Sheet 1 of 2)	Other Revised Figure 8.3.1-3 due to change of the switching method for UPS
8.3-145	Figure 8.3.2-1 (Sheet 1 of 2)	RAI No 670-4773 QUESTION 09.04.05-14 Revised Figure 8.3.1-3 due to clearly state that the battery consists of two strings of 60 cells.
8.4-3	8.4.1.3	Added “Controls exist in the MCR to start, stop and synchronize the AAC power sources.” Reason: Accompanied with Tier-1 review, Tier-2 revision content was reflected.

8.0 ELECTRIC POWER

8.1 Introduction

8.1.1 General

Offsite electric power is provided to the US-APWR plant site from the grid by at least two physically independent transmission lines. ~~The plant's high voltage switchyard is site-specific and is not part of the reference plant design.~~ During the plant startup and shutdown and during all postulated accident conditions, the offsite electric power is supplied to the plant site from the plant high voltage switchyard through two physically independent transmission tie lines. One of these two transmission tie lines connects to the high voltage side of the main transformer (MT), and the other connects to the high voltage side of the reserve auxiliary transformers (RATs). The main generator (MG) is connected to the low voltage side of the MT and the high voltage side of the unit auxiliary transformers (UATs). There is a generator load break switch (GLBS) between the MG and the MT. When the MG is on-line, it provides power to the onsite non safety-related electric power system through the UATs. When the GLBS is open, offsite power to the onsite non safety-related electric power system is provided through the MT and the UATs. With the GLBS either open or closed, offsite power to the onsite safety-related electric power system is provided through the RATs. If power is not available through the UATs, offsite power is provided to both safety-related and non safety-related onsite electric power system through the RATs. Similarly, if power is not available through the RATs, offsite power is provided to both safety-related and non safety-related onsite electric power system through the UATs.

The onsite electric power system provides power to all plant auxiliary and service loads. The onsite electric power system is comprised of alternating current (ac) and direct current (dc) systems. Both ac and dc onsite electric power systems have a safety-related Class 1E power system feeding all Class 1E loads, and a non safety-related non-Class 1E power system feeding all non-Class 1E loads. The Class 1E onsite power system has four independent trains. Each train of the Class 1E ac onsite power system, in addition to their connection to offsite power sources from the grid, has an onsite emergency power source, consisting of a generator driven by a gas turbine. Each train of the Class 1E dc onsite power distribution system, in addition to their connection to corresponding ac train through a battery charger, is provided with a dedicated Class 1E battery power source.

The reference plant has two circuits connected to offsite power sources, four onsite Class 1E emergency gas turbine generator (GTG) power sources, two onsite non-Class 1E GTG power sources and, four Class 1E and four non-Class 1E dc battery power sources. The non-Class 1E GTGs provide power to all electrical loads that are required to bring and maintain the unit in safe-shutdown mode upon the loss of all offsite and onsite ac power sources.

Figure 8.1-1 is a simplified electrical one line diagram depicting the ac and dc onsite and offsite electric power system for the reference plant. The one line diagram containing site-specific information in Section 8.2 is to be provided by the Combined License (COL) applicant.

are electrically isolated and physically separated from all trains of the Class 1E safety-related power distribution system.

8.1.4 Safety-Related Loads

Safety-related loads are defined as those systems and components that require electric power in order to perform their safety functions. The safety-related loads are supplied power from the safety-related Class 1E power distributions systems. The ac safety-related loads are listed in Table 8.3.1-4, Table 8.3.1-7 and Table 8.3.1-10. The dc safety-related loads are listed in Table 8.3.2-1.

8.1.5 Design Bases

8.1.5.1 Offsite Power System

The transmission grid and its interconnections to other generating stations and other grid systems, ~~are site-specific. T~~the plant high voltage switchyard and the transmission tie lines (minimum two) between the plant high voltage switchyard and the transformer yard ~~at the plant site are also site-specific. These site-specific items~~ are discussed in Section 8.2. The following are the design bases that are applicable to offsite power system, irrespective of whether they are part the reference plant design.

All plant loads are supplied offsite power from four UATs or four RATs.

At a minimum, there are two physically independent power circuits between the offsite grid and the plant high voltage switchyard, and between the plant high voltage switchyard and the plant onsite power system. The two power circuits are designed and located to minimize, to the extent practical, the likelihood of their simultaneous failure under operating conditions and postulated accident conditions. Each power circuit has sufficient capacity and capability to assure satisfactory operation of all safety and non safety-related loads.

Upon unit trip for any reason, including a postulated accident, (except due to electrical fault in the power supply circuit affecting the UATs), the GLBS is opened and the plant's non-Class 1E MV buses N1 through N6, P1 and P2 continue to receive power from offsite sources through the UATs. In case of a unit trip due to an electrical fault in the power supply circuit affecting the UATs, the high voltage circuit breaker at the switchyard connected to the MT, and all UAT incoming circuit breakers at the MV switchgear buses N1 through N6, P1 and P2 are opened. MV switchgear buses N1 through N6, P1 and P2 are transferred from the UATs to the RATs. During all modes of plant operation, including startup, normal and emergency shutdown and postulated accident conditions, the Class 1E MV buses A, B, C and D are fed from the normal preferred offsite power source through the RATs. These buses are transferred to the alternate preferred power source through the UATs upon loss of normal preferred power source from the RATs.

8.1.5.2 Onsite Power System

The design bases for the onsite power system are as follows:

- Generic Letter 96-01, “Testing of Safety-Related Circuits.”
- Generic Letter 2006-02, “Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power.”
- Generic Letter 2007-01, “Inaccessible or Underground Power Cable Failures That Disable Accident Mitigation Systems or Cause Plant Transients”

8.1.5.3.5 Institute of Electrical and Electronics Engineers Standards

The ~~offsite and~~ onsite electric power system~~s~~ design conforms to the criteria and recommendations provided in the following IEEE, and other industry standards such as American National Standards Institute (ANSI), National Electrical Manufacturer Association (NEMA), National Fire Protection Association (NFPA) and Insulated Cable Engineers Association (ICEA). The switchyard and transmission system design conforms to the standards discussed in Section 8.2.~~:~~

- IEEE Std 48-1996, “IEEE Standard Test Procedures and Requirements for Alternating-Current Cable Terminations 2.5 kV through 765 kV”
- IEEE Std 80-2000, “IEEE Guide for Safety in AC Substation Grounding”
- IEEE Std 141-1993, “IEEE Recommended Practice for Electric Power Distribution for Industrial Plants”
- IEEE Std 142-2007, “IEEE Recommended Practice for Grounding of Industrial and Commercial Power Systems”
- IEEE Std 242-2001, “IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems”
- IEEE Std 308-2001, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations”
- IEEE Std 317-1983, “IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Generating Stations”
- IEEE Std 323-2003, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations”
- IEEE Std 334-1994, “IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations”
- IEEE Std 336-2005, “IEEE Guide for Installation, Inspection, and Testing for Class 1E Power, Instrumentation, and Control Equipment at Nuclear Facilities”
- IEEE Std 338-2006, “IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems”
- IEEE Std 344-2004, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations”
- IEEE Std 379-2000, “IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems”
- IEEE Std 382-2006, “IEEE Standard for Qualification of Safety Related Actuators for Nuclear Generating Stations”

- IEEE Std 383-2003, “IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations”
- IEEE Std 384-1992, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits”
- IEEE Std 386-2006, “IEEE Standard for Separable Insulated Connector Systems for Power Distribution Systems Above 600 V”
- IEEE Std 387-1995, “IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.” Note: The reference plant has GTG as standby power supply and this standard is applicable to diesel-generator. The criteria and recommendations of the standard that are applicable to GTG are implemented in the standby power supply design.
- IEEE Std 399-1997, “IEEE Recommended Practice for Power System Analysis”
- IEEE Std 420-2001, “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels and Racks Used in Nuclear Power Generating Stations”
- IEEE Std 422-1986, “Guide for the Design and Installation of Cable Systems in Power Generating Stations”
- IEEE Std 434-2006, “IEEE Guide for Functional Evaluation of Insulation Systems for AC Electric Machine Rated 2300 V and Above”
- IEEE Std 446-1995, “IEEE Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications"
- IEEE Std 450-2002, “IEEE Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications”
- IEEE Std 484-2002, “IEEE Recommended Practice for Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations”
- IEEE Std 485-1997, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications”
- IEEE Std 497-2002, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations”
- IEEE Std 505-1977, “IEEE Standard Nomenclature for Generating Station Electric Power Systems”
- ~~IEEE Std 518-1982, “IEEE Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources”~~
- IEEE Std 519-1992, “IEEE Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems”
- ~~IEEE Std 524a-1993, “IEEE Guide to Grounding During the Installation of Overhead Transmission Line Conductors”~~
- ~~IEEE Std 525-2007, “IEEE Guide for the Design and Installation of Cable Systems in Substations”~~
- IEEE Std 535-2006, “IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations”

- IEEE Std 944-1986, "IEEE Recommended Practice for the Application and Testing of Uninterruptible Power Supplies for Power Generating Stations"
- IEEE Std 946-2004, "IEEE Recommended Practice for the Design of dc Auxiliary Power Systems for Generating Stations"
- ~~IEEE Std 980-1994, "IEEE Guide for Containment and Control of Oil Spills in Substations"~~
- ~~IEEE Std 998-1996, "IEEE Guide for Direct Lightning Stroke Shielding of Substations"~~
- IEEE Std 1015-2006, "IEEE Recommended Practice for Applying Low Voltage Circuit Breakers Used in Industrial and Commercial Power Systems"
- IEEE Std 1023-2004, "IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities"
- IEEE Std 1050-2004, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations"
- IEEE Std 1082-1997, "Guide for Incorporating Human Action Reliability Analysis for Nuclear Power Generating Stations"
- IEEE Std 1143-1994, "IEEE Guide on Shielding Practice for Low Voltage Cables"
- ~~IEEE Std 1159.3-2003, "Recommended Practice for the Transfer of Power Quality Data"~~
- IEEE Std 1184-2006, "IEEE Guide for Batteries for Uninterruptible Power Supply Systems"
- IEEE Std 1185-1994, "IEEE Guide for Installation Methods for Generating Station Cables"
- IEEE Std 1202-2006, "IEEE Standard for Flame-Propagation Testing of Wire and Cable"
- IEEE Std 1205-2000, "IEEE Guide for Assessing, Monitoring, and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations" (Corrigendum 1: 2006)
- ~~IEEE Std 1247-2005, "IEEE Standard for Interrupter Switches for Alternating Current, Rated Above 1,000 V"~~
- IEEE Std 1290-1996, "IEEE Guide for Motor Operated Valve (MOV) Motor Application, Protection, Control, and Testing in Nuclear Power Generating Stations"
- IEEE Std 1349-2001, "IEEE Guide for Application of Electric Motors in Class I, Division 2 Hazardous (Classified) Locations"
- IEEE Std 1375-1998, "IEEE Guide for Protection of Stationary Battery Systems"
- ANSI/IEEE 1584-2002, "Guide for Performing Arc Flash Hazard Calculations"
- IEEE Std 1584a-2004, "IEEE Guide for Performing Arc-Flash Hazard Calculations – Amendment 1"

- IEEE Std C37.010-1999, "IEEE Application Guide for AC High Voltage Circuit Breakers Rated on a Symmetrical Current Basis"
- IEEE Std C37.011-2005, "IEEE Application Guide for Transient Recovery Voltage for AC High-Voltage Circuit Breakers"
- ~~IEEE Std C37.013-1997, "IEEE Standard for AC High-Voltage Generator Circuit Breakers Rated on a Symmetrical Current Basis"~~
- IEEE Std C37.2-1996, "IEEE Standard Electrical Power System Device Function Numbers and Contact Designations"
- IEEE Std C37.04-1999, "IEEE Standard Rating Structure for AC High Voltage Circuit Breakers"
- ANSI/IEEE C37.04a-2003, "Standard Capacitance Current Switching Requirements for High Voltage Circuit Breakers"
- IEEE Std C37.06-2000, "American National Standard AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis- Preferred Ratings and Related Required Capabilities"
- IEEE Std C37.13-1990, "IEEE Standard for Low-Voltage AC Power Circuit Breakers Used in Enclosures"
- IEEE Std C37.14-2002, "IEEE Standard for Low-Voltage DC Power Circuit Breakers Used in Enclosures"
- IEEE Std C37.16-2000, "American National Standard Low-Voltage Power Circuit Breakers and AC Power Circuit Breakers – Preferred Ratings, Related Requirements, and Application Recommendations"
- IEEE Std C37.17-1997, "Trip Devices for AC and General Purpose DC Low-Voltage Power Circuit Breakers"
- IEEE Std C37.18-1979, "IEEE Standard Enclosed Field Discharge Circuit Breakers for Rotating Electric Machinery"
- ANSI/IEEE C37.20.1-2002, "Standard for Metal-Enclosed Low-Voltage Power Circuit Breaker Switchgear"
- IEEE Std C37.20.1A-2005, "IEEE Standard for Metal-Enclosed Low-Voltage Power Circuit Breaker Switchgear – Amendment 1: Short-Time and Short-Circuit Withstand Current Tests – Minimum Areas for Multiple Cable Connections"
- IEEE Std C37.20.2-1999, "Metal-Clad and Station-Type Cubicle Switchgear"
- ~~IEEE Std C37.20.3-2001, "Standard for Metal-Enclosed Interrupter Switchgear"~~
- IEEE Std C37.20.4-2001, "Standard for Indoor AC Switches (1 kV to 38 kV) for Use in Metal-Enclosed Switchgear"
- IEEE Std C37.20.6-1997, "IEEE Standard for 4.76 kV to 38 kV Rated Grounding and Testing Devices Used in Enclosures"
- IEEE Std C37.21-2005, "IEEE Standard for Control Switchboards"
- IEEE Std C37.22-1997, "Preferred Ratings and Related Required Capabilities for Indoor AC Medium Voltage Switches Used in Metal-Enclosed Switchgear"

- IEEE Std C37.23-2003, "IEEE Standard for Metal-Enclosed Bus"
- IEEE Std C37.26-2003, "IEEE Guide for Methods of Power Factor Measurement for Low-Voltage Inductive Test Circuits"
- IEEE Std C37.27-1987, "IEEE Standard Application Guide for Low-Voltage AC Non-Integrally Fused Power Circuit Breakers (Using Separately Mounted Current-Limiting Fuses"
- ~~IEEE Std C37.29-1981, "IEEE Standard for Low-Voltage AC Power Circuit Protectors Used in Enclosures"~~
- ~~IEEE Std C37.30-1997, "IEEE Standard Requirements for High Voltage Switches"~~
- ~~IEEE C37.32-2002, "High-Voltage Switches, Bus Supports, and Accessories – Schedules of Preferred Ratings, Construction Guidelines and Specifications"~~
- ~~IEEE Std C37.46-2000, "American National Standard for High Voltage Expulsion and Current-Limiting Type Power Class Fuses and Fuse Disconnecting Switches"~~
- ~~IEEE Std C37.47-2000, "American National Standard for High Voltage Current-Limiting Type Distribution Class Fuses and Fuse Disconnecting Switches"~~
- ~~IEEE Std C37.61-1973, "IEEE Standard Guide for the Application, Operation, and Maintenance of Automatic Circuit Reclosers"~~
- ~~IEEE Std C37.73-1998, "Standard Requirements for Pad-Mounted Fused Switchgear"~~
- IEEE Std C37.81-1989, "IEEE Guide for Seismic Qualification of Class 1E Metal-Enclosed Power Switchgear Assemblies"
- IEEE Std C37.82-1987, "IEEE Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Power Generating Stations"
- IEEE Std C37.90-2005, "IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus"
- IEEE Std C37.90.1-2002, "IEEE Standard for Surge Withstand Capability (SWC) Tests for Relays and Relay Systems Associated with Electrical Power Apparatus"
- IEEE Std C37.90.2-2004, "IEEE Standard for Withstand Capability of Relay Systems to Radiated Electromagnetic Interference from Transceivers"
- IEEE Std C37.91-2000, "IEEE Guide for Protective Relay Applications to Power Transformers"
- IEEE Std C37.98-1987, "IEEE Standard Seismic Testing of Relays"
- IEEE Std C37.105-1987, "IEEE Standard for Qualifying Class 1E Protective Relays and Auxiliaries for Nuclear Power Generating Stations"
- IEEE Std C37.106-2003, "IEEE Guide for Abnormal Frequency Protection for Power Generating Plants"
- IEEE Std C37.121-1989 (R2000), "Switchgear – Unit Substations – Requirements"

- ~~IEEE Std C37.122-1993, "IEEE Standard for Gas-Insulated Substations"~~
- ~~IEEE Std C37.123-1996, "IEEE Guide to Specifications for Gas-Insulated, Electric Power Substation Equipment"~~
- IEEE Std C57.12.00-2000, "IEEE Standard for Standard General Requirements for Liquid-Immersed Distribution, Power and Regulating Transformers"
- IEEE Std C57.13-1993, "IEEE Standard Requirements for Instrument Transformers"
- IEEE Std C57.105-1978, "IEEE Guide for Application of Transformer Connections in Three Phase Distribution Systems"
- IEEE Std C57.109-1993, "IEEE Guide for Transformers Through-Fault Current Duration"
- IEEE Std C62.23, 1995, "IEEE Application Guide for Surge Protection of Electric Generating Plants"
- ANSI/IEEE C2-2002, National Electrical Safety Code
- NEMA MG-1, 2006, Motors and Generators
- NEMA VE-1, 2002, Metal Cable Tray Systems
- NFPA 70-2005, National Electrical Code
- NFPA 780-2004, Standard for the Installation of Lightning Protection Systems
- ICEA P-54-440/NEMA WC-51, 2003, Ampacities of Cables Installed in Cable Trays

8.1.6 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

8.1.7 References

8.1-1 IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, IEEE Std 387, 1995.

8.1-2 Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants, Regulatory Guide 1.9 Rev. 4, March 2007.

8.1-3 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix A.

Table 8.1-1 Design Criteria and Guidelines for Electric Power Systems (Sheet 2 of 7)

Criteria Provided in Referenced Documents	DCD Section/Subsection				Remarks
	8.2	8.3.1	8.3.2	8.4	
2. Regulations (10 CFR 50 and 10 CFR 52)					
a. 10 CFR 50.34, “Contents of Applications; Technical Information”					
i. 50.34(f)(2)(v) (Related to TMI Item I.D.3)	~~A~~	~~A~~	~~A~~		Not applicable
ii. 50.34(f)(2)(xiii) (Related to TMI Item II.E.3.1)		~~A~~			Not applicable
iii. 50.34(f)(2)(xx) (Related to TMI Item II.G.1)		~~A~~			Not applicable
b. 10 CFR 50.55a, “Codes and Standards”		A	A		
c. 10 CFR 50.63, “Loss of All Alternating Current Power”	A	A	A	A	
d. 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”	A	A	A	A	
e. 10 CFR 52.47(b)(1), “Contents of Applications”	A	A	A	A	
f. 10 CFR 52.80(a), “Contents of Applications; Additional Technical Information”	A	A	A	A	

Note: “A” denotes that US-APWR conforms to the requirements and criteria provided in the subject document.
“G” denotes that US-APWR conforms to the guidance provided in the subject document.

8.2 Offsite Power System

8.2.1 System Description

8.2.1.1 Transmission System

The transmission system is not within the scope of the US-APWR; however, there are design basis requirements (interface requirements) which are described in Subsection 8.2.3. The COL applicant is to implement the transmission system interfaces for the US-APWR.

8.2.1.2 Offsite Power System

The offsite power system is a non safety-related, non-Class 1E system. The plant switchyard is connected to the transmission grid by at least two electrically independent and physically isolated power circuits. As a minimum, there are two electrically isolated and physically independent power circuits connecting the plant to the switchyard. The COL applicant is to assure at least two electrically isolated and physically independent power circuits as normal and alternate preferred power sources.

Offsite electric power is provided to the onsite power system from the grid and other generating stations by at least two physically independent transmission lines. The plant's high voltage switchyard is site-specific and not a part of the reference plant design. During plant startup, shutdown, maintenance, and during all postulated accident conditions, offsite electric power can be supplied to the plant site from the plant high voltage switchyard through two physically independent transmission tie lines. One of these two transmission tie lines connects to the high voltage side of the MT, and the other connects to the high voltage side of the RATs. The MG is connected to the low voltage side of the MT and the high voltage side of the UATs. There is a GLBS between the MG and the MT. When the MG is on-line, it provides power to the onsite non safety-related electric power system through the UATs. When the GLBS is open, offsite power to the onsite non safety-related electric power system is provided through the MT and the UATs. With GLBS either open or closed, offsite power to the onsite safety-related electric power system is provided through the RATs. If power is not available through the UATs, offsite power is provided to both safety-related and non safety-related onsite electric power systems through the RATs. Similarly, if power is not available through the RATs, offsite power is provided to both safety-related and non safety-related onsite electric power system through the UATs. Both normal and alternate preferred power sources have the capability to serve the total plant auxiliary and service loads during all modes of plant operation including postulated accident conditions.

The MG is connected to the GLBS through an ~~air cooled~~ isolated phase busduct cooled by forced air with water cooling. The other side of the GLBS is connected to the low voltage side of the MT, also through an ~~air cooled~~ isolated phase busduct. This isolated phase busduct has a tap connection to the high voltage side of the UATs through a disconnect link. The MT consists of three single phase transformers with one installed spare.

There are two non-Class 1E 13.8 kV MV buses N1 and N2, four non-Class 1E 6.9 kV MV buses N3, N4, N5 and N6, two non-Class 1E 6.9 kV MV permanent buses P1 and

P2 and four Class 1E 6.9 kV MV buses A, B, C and D. MV bus N1 can be fed from UAT1 or RAT1. MV bus N2 can be fed from UAT2 or RAT2. MV buses N3, N4, A, B and P1 can be fed from UAT3 or RAT3. MV buses N5, N6, C, D and P2 can be fed from UAT4 or RAT4. For all these MV buses, if power is lost from one source, the buses are automatically transferred to the other source by fast or slow transfer scheme. At that time, if bus voltage is adequate, fast transfer is initiated. If this is not the case, slow transfer is initiated. Performance of these transfers is permitted when the bus faulted signal is not initiated. Detailed explanation of bus transfer scheme is described in Subsection 8.3.1.1.2.4. All low voltage buses are provided power from the MV buses. Each of the 6.9 kV Class 1E MV buses has its own onsite Class 1E standby emergency power source. Similarly, each of the 6.9 kV non-Class 1E MV permanent buses has its own onsite non-Class 1E standby emergency power source, designated as AAC power source. All MV buses can be powered from their associated UAT or RAT.

There are four, two-winding, UATs, namely UAT1, UAT2, UAT3, and UAT4. The high-side of these transformers is connected to the main generator isolated phase busduct down-stream of the GLBS. During normal power operation, with the GLBS closed, the MG provides power to the plant MV buses N1, N2, N3, N4, N5, ~~and~~ N6 P1 and P2 through the UATs. During all other modes of plant operation, including PAs, with the GLBS open, these MV buses are powered through the UATs by back-feeding the MT from the offsite power sources. During all modes of plant operation including startup, normal and emergency shutdown and PAs, the MV Class 1E buses A, B, C and D are powered through the RATs from offsite power sources. Secondary voltages of UAT and RAT are displayed in the MCR.

There are four, three-winding RATs, namely RAT1, RAT2, RAT3, and RAT4. The high-side of these transformers is connected to the high voltage transmission tie line from the switchyard. The transmission tie line voltage level is site-specific. This is the normal preferred power source for all plant safety-related auxiliary and service loads. RAT1 and RAT2 can feed the 13.8 kV non-Class 1E buses N1 and N2, respectively. RAT3 can feed the 6.9 kV Class 1E buses A and B, and non-Class 1E buses N3, N4 and P1. RAT4 can feed the 6.9 kV Class 1E buses C and D, and non-Class 1E buses N5, N6 and P2.

Each of the safety-related and non safety-related MV buses (13.8 kV non-Class 1E buses N1 and N2; 6.9 kV Class 1E buses A, B, C, and D; and 6.9 kV non-Class 1E buses N3, N4, N5, N6, P1 and P2) is connected to a UAT and an RAT. For all Class 1E (A, B, C and D) MV buses, power from the RAT is the normal preferred source and power from the UAT is the alternate preferred source. Each safety MV bus also has its own backup emergency power supply from a safety-related Class 1E GTG. MV permanent buses P1 and P2 also have their own backup emergency power supply from a dedicated non-Class 1E GTG.

During all modes of plant operation, including normal and emergency shutdown and postulated accident conditions, all safety-related unit auxiliary and safety-related plant service loads are powered from offsite power sources through the RATs. This is the normal preferred offsite power source for the plant safety-related loads. The alternate preferred offsite power source to the plant safety-related loads is from the UATs, which are powered from offsite power sources by back feeding the MT. All plant MV buses, both safety-related and non safety-related, are connected to the UATs and RATs through bus incoming circuit breakers. If power to any MV bus is lost from the normal source, it is automatically transferred to the alternate source. If any one UAT becomes inoperable, it can be isolated from the system and the affected MV buses can be powered from the backup RAT.

During a coincident loss of offsite power (LOOP) and loss-of-coolant accident (LOCA), the safety-related MV buses are powered from onsite Class 1E emergency GTG power sources. The unit is also provided with alternate ac power sources for powering the loads that are needed to operate during a station blackout (SBO) event. The equipment and circuits that are associated with the offsite power system are physically independent from the onsite power system and the alternate ac sources. Any single failure in the offsite power system, in the onsite power system, or in the AAC sources will have no impact on the availability of the remaining systems.

The main transformers, UATs and RATs are designed and constructed to withstand mechanical and thermal stresses produced by the worst-case external short circuit, and meet the corresponding requirements of IEEE Std C57.12.00 (Reference 8.2-1).

The ratings of the MG, the GLBS, the MT, the UATs and the RATs are as follows:

Equipment	Rating
Main generator	1,900 MVA, 26 kV, 60 Hz
Generator load break switch	~~27.3~~28 kV, 44.4 kA, 60 Hz
Main transformer	Three single phase transformers and one installed spare, each 610 MVA for a combined rating of 1,830 MVA, 60 Hz, low voltage side is 26 kV, high voltage side is site-specific.
Unit auxiliary transformers (UAT1 and UAT2)	72 MVA, 26 – 13.8 kV, 60 Hz
Unit auxiliary transformers (UAT3 and UAT4)	53 MVA, 26 – 6.9 kV, 60 Hz
Reserve auxiliary transformers (RAT1 and RAT2)	72 MVA, 60 Hz, low voltage side 13.8 kV, high voltage side is site-specific.
Reserve auxiliary transformers (RAT3 and RAT4)	53 MVA, 60 Hz, low voltage side 6.9 kV, high voltage side is site-specific.

The GLBS has the capability to break the maximum credible generator full load current. For normal or emergency plant shutdown for all design-basis events (DBE) except for an electrical fault in the 26 kV power system or associated equipment and circuits, the GLBS is opened and power to all auxiliary and service loads is maintained without any interruption from the alternate preferred offsite power source through the UATs. During emergency shutdown of the plant due to any electrical fault in the 26 kV system or associated equipment and circuits, the fault is isolated by opening the main circuit breaker on the high voltage side of the MT and all incoming circuit breakers of the MV buses connected to the UAT power source; and all affected MV buses are automatically transferred to the RAT source. The MV Class 1E ~~and non-Class 1E P buses~~ are not affected since these are normally fed from the RATs. The UAT incoming breakers to these buses is locked out and blocked from closing.

Unit synchronization is normally through the GLBS. Synchrocheck relays are used to ensure proper synchronization of the unit to the offsite power system.

High voltage circuit breakers are sized and designed in accordance with IEEE Std C37.010 and C37.06 (Reference 8.2-14, 8.2-15). High voltage disconnecting switches are sized and designed in accordance with IEEE Std C37.32 (Reference 8.2-16).

The MTs, UATs and RATs have differential, over-current, sudden pressure and ground over-current protection schemes per IEEE Std 666 (Reference 8.2-9). The COL applicant is to provide site-specific protection scheme.

Isolated phase busduct provides the electrical interconnections between generator load terminals to the GLBS, the GLBS to the MT and the disconnect links on the high voltage side of the UATs, and the UAT disconnect links to the UATs. Non-segregated phase bus ducts/cable buses provide electrical connections between the low voltage side of the UATs and RATs to the 13.8 kV and 6.9 kV MV switchgear. The non-segregated phase bus ducts/cable buses from the UATs and RATs are physically separated to minimize the likelihood of simultaneous failure.

Each of the single phase transformers of the MT is provided with disconnect links so that a failed transformer may be taken out of service and the spare transformer can be connected. All UATs are also provided with disconnect links so that a failed transformer can be taken out of service. With one UAT or one RAT out of service, all MV buses will have access to at least one offsite power source.

The MT, UATs and RATs, are located in the area of MT and UATs and area of RAT, respectively, separated by three-hour rated fire barriers, in the transformer yard adjacent to the turbine building (T/B). Cables associated with the normal preferred and alternate preferred circuits are physically separated from each other to minimize common cause failure[[, even supposing that these circuits share a common underground duct bank]]. In accordance with the guidance of Generic Letter 2007-01, for preventing the degradation of medium voltage cables that are installed in underground duct banks, the manholes are at the low point with the conduits in the connecting duct banks sloped for water drainage into the manholes. The manholes are available for temporary sump pumps for water draining. The medium voltage cables whether in a duct bank or in a conduit are monitored by periodical testing, such as partial discharge testing, time domain reflectometry, dissipation factor testing, and very low frequency AC testing. These PPS circuits are also designed to minimize common cause failure with standby power sources. All of these transformers are provided with containment for collection of transformer oil in case of tank leakage or rupture.

8.2.1.2.1 Switchyard

This RG endorses Section 11 of NUMARC 93-01 (Reference 8.2-6) dated February 11, 2000 with some provisions and clarifications for complying with 10 CFR 50.65(a)(4) (Reference 8.2-7). Conformance to this regulatory guide is generically addressed in Section 1.9.

- RG 1.204, " Guidelines for Lightning Protection of Nuclear Power Plants"

  This RG endorses four IEEE Standards, IEEE Std 665 (Reference 8.2-8), IEEE Std 666 (Reference 8.2-9), IEEE Std 1050 (Reference 8.2-10) and IEEE Std C62.23 (Reference 8.2-11), in their entirety with one exception to IEEE Std 665 (Reference 8.2-8), Subsection 5.7.4, which misquotes subsection 4.2.4 of IEEE Std 142 (Reference 8.2-12). The US-APWR offsite power supply design fully conforms to the requirements of the endorsed IEEE standards that pertain to the lightning protection of nuclear power plants.

- Standard Review Plan (SRP) Section 8.2 Appendix A

  The US-APWR has GLBS which ~~is designed and tested in accordance with~~ conforms to the requirement of the SRP Section 8.2 Appendix A. In addition, the Class 1E MV buses are normally supplied from RAT as normal PPS. Therefore, immediate access circuit is assured without isolating the main generator from MT and UAT in case of electrical fault in the power supply circuit affecting the UATs.

- BTP 8-3, "Stability of Offsite Power Systems"

  This topic is site-specific (see Subsection 8.2.4, Combined License Information).

- BTP 8-6 "Adequacy of Station Electric Distribution System Voltages"

  US-APWR design provides second level of undervoltage protection with time delays to protect the Class 1E equipment from sustained undervoltages.

8.2.3 Design Bases Requirements

The offsite power system of the US-APWR reference plant is based on certain design bases (as defined in 10 CFR 50.2 (Reference 8.2-13)) requirements.

The COL applicant is to provide failure modes and effects analysis (FMEA) of offsite power system for conformance with following requirements.

- The normal and alternate preferred power supply circuits originating from separate transmission substations connect to the onsite ac power system, through the plant switchyard(s). Both circuits may share a common switchyard. The normal preferred circuit and the alternate preferred circuit are electrically isolated and physically independent from each other to the extent practical to minimize common mode failure. Each circuit is capable of supplying all unit auxiliary and service loads during normal plant power operation, as well as during normal or emergency shutdown.
- In case of failure of the normal preferred power supply circuit, the alternate preferred power supply circuit remains available.

8.3.1.1.1 Non-Class 1E Onsite AC Power System

The 13.8kV ac system includes non-Class 1E buses N1 and N2. Bus N1 is connected to either UAT1 or RAT1 and bus N2 is connected to either UAT2 or RAT2 by non-segregated busduct/cable buses. The ratings of UAT1, UAT2, RAT1 and RAT2 are shown in Table 8.3.1-1.

The non-Class 1E 6.9kV ac system includes buses N3, N4, N5, N6, P1 and P2. Buses N3, N4 and P1 are connected to either UAT3 or RAT3 and buses N5, N6 and P2 are connected to either UAT4 or RAT4 by non-segregated busduct/cable buses. The ratings of UAT3, UAT4, RAT3 and RAT4 are shown in Table 8.3.1-1.

The UAT and RAT ratings are adequate to meet the maximum load requirements during normal plant operation, start-up, shutdown and design-basis events, as shown in Table 8.3.1-3.

Non-Class 1E 6.9kV permanent buses P1 and P2 are also connected to the non-Class 1E A-AAC GTG and B-AAC GTG, respectively. The loads which are not safety-related but require operation during LOOP are connected to these buses. The AAC GTGs are of different rating and are provided with diverse starting mechanisms as compared to the Class 1E GTGs. The AAC GTGs are selected as non-Class 1E to minimize common cause failures with the Class 1E GTGs. The AAC GTGs are started by dc supplied from batteries and the Class 1E GTGs are started by a compressed air system. Rating of AAC GTG is shown in Table 8.3.1-1. Any one AAC GTG is adequate to meet the load requirements shown in Table 8.3.1-5 and Table 8.3.1-6 during LOOP and SBO conditions.

Normal offsite power to the non-Class 1E 13.8kV buses N1, N2 and non-Class 1E 6.9kV buses N3, N4, N5, N6, P1 and P2 is provided from the UATs and alternate offsite power is provided from the RATs. Automatic bus transfer schemes are provided on all these buses to automatically transfer the loads from the normal offsite power source to the alternate offsite power source in case of loss of normal power to the buses.

Logic schemes for the automatic fast and slow bus transfer of offsite power from UAT to RAT for non-Class 1E MV buses N1, N2, N3, N4, N5 ~~and~~ N6, P1 and P2 ~~and RAT to UAT for non-Class 1E MV buses P1 and P2~~ are shown in Figure 8.3.1-2. Restoration of power from the alternate offsite source back to normal offsite source is by manual operation.

LOOP condition occurs if power from both the UAT and RAT is lost to the onsite ac power system buses. Motor loads fed from these buses are tripped by the bus undervoltage relays. However, power to the non-Class 1E 6.9kV ac permanent buses P1 and P2 is automatically restored from the A-AAC GTG and B-AAC GTG respectively. The A-AAC GTG is started automatically by the undervoltage relays on bus P1 and B-AAC GTG is started automatically by the bus undervoltage relays on bus P2 during the LOOP condition. As soon as the AAC GTGs reach their preset voltage and frequency limits, the circuit breakers connecting the A-AAC GTG and B-AAC GTG to their respective selector circuits A and B are closed, as shown in Figure 8.3.1-2. The circuit breakers in the 6.9kV switchgears P1 and P2 and the disconnect switches in the selector switches A and B, connecting the 6.9kV bus P1 to selector circuit A and bus P2 to administratively controlled and are closed manually during SBO condition. Class 1E 6.9kV buses A or B can be connected to the A-AAC GTG, and Class 1E 6.9kV buses C or D can be connected to the B-AAC GTG, during SBO condition. The major Class 1E distribution equipment of train A, B, C and D are physically separated by different rooms as shown in Figure 8.3.1-4. Redundant safe shutdown components and associated redundant Class 1E electrical trains are separated from the other Class 1E and non-Class 1E systems by 3-hour rated fire barriers to preserve the capability to safely shutdown the plant following a fire (Subsection 9.5.1.1). Access to the Class 1E power equipment areas is administratively controlled. The R/B and safety-related PS/Bs are structurally designed to meet seismic category I requirements as defined in RG 1.29 (Reference 8.3.1-3). These structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform safety functions. They are also designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions (Subsection 1.2.1.2.11). The orientation of the R/B and safety-related PS/Bs where Class 1E onsite power system components are located, is such that the probability of a turbine missile striking the R/B or PS/Bs is minimum. The Class 1E onsite power system components are also protected from internally generated missiles and tornado generated missiles. Safety-related components are protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids as a result from equipment failure or events and conditions outside the nuclear power unit. Class 1E equipment important to safety will be protected from the dynamic effects of pipe rupture and are capable of performing their intended safety functions (Subsection 3.6.1).

Auxiliary support systems such as fuel oil systems, compressed air systems and control power supplies are also separate and independent for each Class 1E train. The Class 1E power to the auxiliary support systems is derived from the same train they serve. The heating, ventilation, and air conditioning (HVAC) systems that support operation of the Class 1E ac distribution equipment are powered from the redundant Class 1E ac power system as described in Subsection 9.4.5.

The four Class 1E trains are electrically isolated from the offsite power supplies and each other. The power sources to Class 1E 6.9kV buses are not operated in parallel except for a short period of time during the testing of Class 1E GTGs in parallel with an offsite source. There are no automatic tie connections between the redundant Class 1E trains. The manual tie connection between train B load center and train A load center A1, and between train C load center and train D load center D1 are closed manually, only during the maintenance of the Class 1E A-GTG or Class 1E D-GTG. The tie circuit breakers are mechanically interlocked to prevent parallel connection of load center A1 to load centers A and B, and load center D1 to load centers C and D.

Non-Class 1E loads, except for the emergency lighting and pressurizer heater circuits, are not supplied from the Class 1E power systems. The circuits for non-Class 1E loads are electrically isolated from the Class 1E power system by Class 1E isolation devices. Pressurizer heater Back-up groups are supplied from the Class 1E power systems based on 10CFR50.34(f) (2) (xiii).

a. Demonstrate that on loss of offsite power the Class 1E buses are de-energized and that the loads are shed from the Class 1E buses in accordance with design requirements.

b. Demonstrate that on loss of offsite power the Class 1E GTGs start on the auto start signal, load shed occurs, the Class 1E buses are energized along with SSTs, the auto connected shutdown loads are energized through the load sequencer, and the system operates for 5 minutes while the Class 1E buses are loaded with the shutdown loads.

c. Demonstrate that on an engineered safety features actuation signal (without loss of offsite power) the Class 1E GTGs start on the auto start signal and operate on standby for 5 minutes.

d. Demonstrate that on loss of offsite power, in conjunction with an engineered safety features actuation signal, the Class 1E GTGs start on the auto start signal, load shedding occurs, the emergency buses are energized along with SSTs, the auto connected emergency (accident) loads are energized through the load sequencer, and the system operates for 5 minutes while the Class 1E GTGs are loaded with the emergency loads.

e. Demonstrate ~~maximum expected~~ load-carrying capability for 24 hours, of which 22 hours are at a load equivalent to ~~the~~ 90 – 100 % of the continuous rating~~maximum expected loading~~ of the Class 1E GTG and 2 hours at a load equivalent to or greater than 105 % of the ~~maximum expected loading~~continuous rating of the Class 1E GTG.

f. Demonstrate functional capability at full load temperature conditions by verifying the Class 1E GTG starts upon receipt of a manual or auto-start signal, and the generator voltage and frequency are attained within the required time limits.

g. Demonstrate proper operation during Class 1E GTG load shedding, including a test of the loss of the largest single load and of complete loss of load. Verify that the overspeed limit is not exceeded.

h. Demonstrate the ability to:

- Synchronize the Class 1E GTG unit with the offsite system while the unit is connected to the emergency load.
- Transfer the emergency load to the offsite system.
- Restore the Class 1E GTG to standby status.

i. Demonstrate that the fuel transfer pumps transfer fuel from each fuel storage tank to the day tank of each Class 1E GTG.

j. Demonstrate that, with the Class 1E GTG operating in a test mode and connected to its Class 1E bus, a simulated ECCS signal overrides the test mode by: (1)

1. The A MOV MCC1, A MOV MCC2, B MOV MCC, C MOV MCC, D MOV MCC1 and D MOV MCC2 are normally fed from the corresponding train of the MOV inverter each of which is backed up by the pertinent Class 1E 125V dc bus as shown in Figure 8.1-1. The Class 1E ac MCC backups the associated train MOV MCC in case of loss of MOV inverter output.

In the event of a postulated LOCA and coincident LOOP, the battery charger input power to the MOV inverters may be lost for up to 100 seconds until the onsite Class 1E GTGs are ready to accept loads. Each MOV is started at the required time by automatic starting signal for equalization of dc current. The 125V dc batteries and the MOV inverters are sized for continuous operating load and coincident starting load of all MOVs actuated by an engineered safety features actuation signal. The MOVs that are required to operate by an engineered safety features actuation signal and their load currents are shown in Table 8.3.1-10.

8.3.1.1.6 Class 1E 120V AC I&C Power Supply

There are four independent Class 1E 120V ac I&C power supply trains A, B, C & D to supply four trains of the protection and reactor control systems, as shown in Figure 8.3.1-3. Each train consists of an UPS, a bypass transformer, a switching circuit and 120V ac distribution panels. Input to the UPS and the bypass transformers in each train is obtained from the ac and dc buses belonging to the same train.

UPS units A, B, C and D are connected to 120V ac distribution panels A, B, C and D respectively through the switching circuit.

The bypass transformers A, B, C and D are also connected to the 120V ac distribution panels A, B, C and D respectively through the switching circuit.

Normally 120V ac distribution panels A, B, C and D are fed from the 50kVA, 1 phase UPS units A, B, C and D respectively. Each UPS and bypass transformer is connected to the 120V ac distribution panel. Each bypass transformer is connected to the 120V ac bus via switching circuit when the inverter is not in service. Normally the 120V ac bus is fed from the inverter. When the inverter fails or is out on maintenance, the 120V ac bus is transferred to the bypass transformer by static switch or manually through synchronizing, without interruption of power to the loads. The static bypass switch has the capability of automatically retransferring the load back to the inverter after its output has returned to normal.~~In case of failure of the UPS unit or if the UPS unit is out on maintenance, buses A, B, C and D are switched to the 50kVA, 480V/120V ac bypass transformer associated with the same train. Switching circuits are provided with contactors for transfer between the UPS unit power and the transformer power. When the input power of switching circuit from UPS unit is lost, undervoltage relay actuates. Following the undervoltage signal, a contactor of UPS unit side is opened, and then a contactor of transformer side is closed with time delay, automatically. Administrative controls ensure that no more than one vital ac bus is powered from the bypass transformer at any time during routine preventive maintenance of the associated UPS unit. The transfer from the transformer back to the UPS is performed manually.~~

During LOOP input to the UPS unit is powered by the Class 1E battery and the supply to the 120V ac distribution panel is restored without interruption.

Output voltage and current of UPS and transformer are displayed in the MCR. Position of switching circuit and voltage of buses are also displayed in the MCR.

Protection of UPS is provided in accordance with IEEE-446 and recommendation from manufactures. The fault current, over current, overvoltage and undervoltage are basic protection schemes. In addition an inverter is also commonly supplied with current-limiting capability for protection. Distribution devices are to be coordinated with this inverter’s current-limiting capability.

8.3.1.1.7 Non-Class 1E 120V AC I&C Power Supply

The non-Class 1E 120V ac I&C power supply is designed to furnish reliable power to ~~all~~ non safety-related plant instruments and controls. There are nine non-Class 1E 120V ac I&C power supply systems as shown in Figure 8.3.1-3. Each system consists of a UPS unit with an inverter and a bypass transformer, bypass switches and 120V ac distribution panels. The ac input to the inverter and the bypass transformer is provided from different MCCs connected to 480V ac permanent buses P1 and P2. This arrangement results in the availability of ac power to the inverters from either of the permanent buses P1 or P2.

Each UPS and bypass transformer is connected to the 120V ac distribution panel. Each bypass transformer is connected to the 120V ac bus via switching circuit when the inverter in not in service.

Normally the 120V ac bus is fed from the inverter. When the inverter fails or is out on maintenance, the 120V ac bus is transferred to the bypass transformer by static switch or manually through synchronizing, without interruption of power to the loads.

When a LOOP occurs, the inverter is fed from the battery for the time required for the non-Class 1E I&C loads that include AAC GTG control to start and begin accepting load. The ac input power to the inverter and/or the bypass transformer will be automatically restored.

The non-Class 1E UPS systems are rated for 120V ac, 1 phase, 60kVA units.

8.3.1.1.8 Electrical Equipment Layout

The locations of Class 1E equipment are selected to minimize vulnerability to physical damage. Wherever practicable, electrical equipment is located away from mechanical piping in order to minimize the damaging effects of pipe ruptures. The degree of separation takes into account the potential hazards in a particular area. Separation is achieved by locating equipment and circuits in separate rooms, maintaining distance, or by use of barriers. The potential hazard of non safety-related equipment failure on safety-related redundant equipment is considered in the choice of equipment location or protection.

Class 1E switchgear and equipment located below the probable maximum flood level are protected as described in Section 3.4.

The following are the general features of the electrical equipment layout:

- UATs and RATs are located outdoors, physically separated from each other as shown on Figure 8.2-1.
- Non-Class 1E 13.8kV and 6.9kV switchgears and 480V load centers N1, N2, N3, N4, N5, N6, P1 and P2 are located separately in two T/B electrical rooms.
- Non-Class 1E MCCs P12, P22, N12, N22, N32 and N42 are located separately in ~~two~~ T/B ~~electrical rooms~~area local. Non-Class 1E MCCs P11, P21, N11, N21, N31 and N41 are located in the auxiliary building (A/B).
- Non-Class 1E batteries N1 and N2 are located in A/B. Non-Class 1E batteries N3 and N4 are located separately in two T/B electrical rooms.
- Non-Class 1E UPS units and 120V ac buses N11, N12, N31, N32, N41, N42 and N5; battery chargers N1, N2 and N12, and 125V dc buses N1 and N2 are located in the A/B.
- Non-Class 1E UPS units and 120V ac buses N21 and N22; battery chargers N3, N4 and N34, 125V dc buses N3 and N4 are located in T/B electrical room.

8.3.1.1.9 Design Criteria for Class 1E Equipment

Design criteria for the Class 1E equipment are discussed below.

Motor size

The nameplate horsepower rating of the motor is selected to equal or exceed the maximum horsepower required by the driven load under normal running or runout conditions.

Class 1E motors larger than 300HP are connected to the 6.9kV Class 1E buses. Class 1E motors up to 300HP are connected to the 480V Class 1E load center buses.

Minimum motor accelerating voltage

All Class 1E motors required to be started when GTG is supplying the 6.9kV and 480V Class 1E buses during LOOP condition are specified with accelerating capability to accelerate to full speed at 80% of nameplate voltage for a 6.6kV motor, and at 75% of nameplate voltage for a 460V motor. The voltage drop calculations show that the minimum voltage at the motor terminals during starting is above 80% of nameplate rating for a 6.6kV motor, and at 75% for a 460V motor. The electrical system is designed so that the terminal voltage at each Class 1E motor will permit acceleration of that motor in the required time.

Motor starting torque

The motor starting torque is adequate for starting and accelerating the connected load to normal speed within sufficient time to perform its safety function for all expected operating conditions, including design minimum bus voltages stated in Table 8.3.1-2. In

This RG requires ensuring that safety-related MOVs whose motors are equipped with thermal overload protection devices integral with the motor starter will perform their function. Design of the MOVs overload protection devices is in accordance with this requirements.

- RG 1.118, “Periodic Testing of Electric Power and Protection Systems”

  This regulatory guide endorses IEEE Std 338 (Reference 8.3.1-29) with some exceptions and clarifications indicated in regulatory positions C (1) through C (3). The IEEE Std 338 (Reference 8.3.1-29) provides design and operational criteria for the performance of periodic testing as part of the surveillance program of nuclear plant safety systems. The IEEE Std 338 (Reference 8.3.1-29) was reissued in 2006 and the surveillance program for the Class 1E ac power system conforms to the criteria provided in the 2006 version of the IEEE standard. The regulatory positions cited in RG 1.118 (Reference 8.3.1-24) are of a clarifying nature and the intent of these regulatory positions is considered in developing the periodic testing program for the Class 1E ac power system.

- RG 1.153, “Criteria for Safety Systems”

  This regulatory guide endorses IEEE Std 603 (Reference 8.3.1-4) with some clarifications regarding applicability and use of industry standards referenced in Section 3 of IEEE Std 603 (Reference 8.3.1-4). IEEE Std 603 (Reference 8.3.1-4) provides minimum functional and design requirements for the power, instrumentation, and control portions of safety systems for nuclear power generating stations. The IEEE Std 603 (Reference 8.3.1-4) was reissued in 1998 and this later version of the standard is used to establish the minimum functional and design requirements for the safety dc power system. The regulatory positions cited in RG 1.153 (Reference 8.3.1-5) are of clarifying nature and the intent of these regulatory positions is considered in establishing the minimum functional and design requirements for the safety-related ac power system.

- RG 1.155, “Station Blackout”

  This regulatory guide provides guidance for complying with 10 CFR 50.63 (Reference 8.2-5). The plant has two AAC power sources of which only one is required to be operational to cope with an SBO event. The AAC power source design, operation, testing, maintenance and associated quality assurance requirements conform to the guidance provided in RG 1.155. Power to all electrical loads that are required to be operational, is restored within one hour from the onset of an SBO event. AAC source power to only one Class 1E 6.9kV bus is required to cope with an SBO event. Non-Class 1E equipment and circuits that are associated with the AAC power sources are completely independent from the onsite Class 1E standby power sources and the offsite power sources. The Class 1E GTGs are ~~never~~ not operated in parallel with the AAC GTGs~~.~~ except briefly during recovery from SBO. The AAC GTGs are not operated in parallel with offsite power sources except during testing of AAC GTGs~~.~~ and recovery from SBO. The AAC GTGs are of different size and have different starting system from the Class 1E GTGs.

with all downstream protective devices in accordance with IEEE Std 242 (Reference 8.3.1-39).

8.3.2.2 Analysis

The US-APWR Class 1E 125V dc power system conforms to the requirements of NRC regulations and GDC identified in Table 8.1-1 in Section 8.1. The system design, installation and operation conform to the guidance of RGs and BTPs identified in Table 8.1-1 in Section 8.1. Specifically, the Class 1E 125V dc power system and its components conform to the requirements of GDC 2, 4, 17, 18, 33, 34, 35, 38, 41, 44 and 50 of Appendix A to 10 CFR 50; and the system design, installation, testing and operation are in accordance with regulatory guidance provided in RG 1.6, RG 1.32, RG 1.47, RG 1.53, RG 1.63, RG 1.75, RG 1.118, RG 1.128, RG 1.129, RG 1.153, RG 1.155, RG 1.160 and RG 1.182.

Since the DC systems are stand alone, which each train having its independent battery, battery charger, etc. all switching are manual. As a result no logic diagrams are required.

8.3.2.2.1 Compliance with General Design Criteria

- Criterion 2 – Design bases for protection against natural phenomena

  All equipment and components of the safety-related Class 1E 125V dc power system are located in a seismic category I building and their mounting and installations are seismically designed. The Class 1E 125V dc system is designed to withstand the effects of natural phenomena such as design basis earthquake, tornado, hurricane, flood, tsunami, or seiche without losing its capability to perform their intended safety functions. Compliance to GDC 2 for all safety-related SSCs is generically addressed in Section 3.1.

- Criterion 4 – Environmental and dynamic effect design bases

  All equipment and components of the safety-related 125V dc power system are designed to withstand the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing and postulated accidents; and are appropriately protected against dynamic effects that may result from equipment failures, including missiles. The safety-related dc power system is designed to perform its intended safety functions during normal, abnormal, accident and post-accident conditions. The safety-related 125V dc power system is comprised of four independent trains that are electrically isolated and physically separated. The batteries of each train are located in a separate Class 1E battery room with safety-related ventilation system. The battery charger and the main distribution switchboard of each train are located in the Class 1E battery charger room for the same train, also provided with redundant train safety-related HVAC system. There are no high or moderate energy lines or missile generating rotating equipment in the Class 1E Battery Rooms or Class 1E Battery Charger Rooms. All equipment and components of the safety-related dc system are qualified for Class 1E application in accordance with IEEE Std 323 (Reference 8.3.1-6) and all applicable IEEE equipment qualification standards. Compliance to GDC 4 for all safety-related SSCs is generically addressed in Section 3.1.

complying with 10 CFR 50.65(a)(4) (Reference 8.2-7). Conformance to this regulatory guide is generically addressed in Section 1.9.

8.3.2.3 Electrical Power System Calculations and Distribution System Studies for DC System

Load flow, voltage regulation, short circuit studies and equipment sizing studies are performed following guidance provided in IEEE Std 946 (Reference 8.3.2-1) and other referenced IEEE standards.

8.3.2.3.1 Load Flow and Under-/Overvoltage Protection

Load flow studies are performed to evaluate whether an acceptable voltage range is maintained at the equipment terminal under worst case loading conditions. Voltage drop at equipment terminal is also calculated under maximum discharge conditions. As a result, terminal voltage of the equipment satisfies the acceptable voltage range based on IEEE Std 946 (Reference 8.3.2-1).

8.3.2.3.2 Short Circuit Studies

Short circuit studies are performed to determine the magnitude of the prospective currents flowing throughout the power system due to a fault occurrence. The studies are performed to calculate the most severe fault condition. This condition is a short circuit at the output terminal of a circuit breaker in the onsite dc distribution system. There are no continuously operating dc motors connected to the Class 1E dc systems. Hence, the Class 1E dc bus short circuit calculations considered only batteries and battery-charger contributions to the fault. The acceptance criteria are that the calculated maximum short circuit current conforms to applicable breaker capability. Table 8.3.2-3 shows the breakers nominal ratings for dc power system. The COL applicant is to perform short circuit studies to confirm breaker ratings.

8.3.2.3.3 Equipment Sizing Studies

Table 8.3.2-1 shows dc spreadsheet load lists. The dc power system equipment sizing is based on the list. The battery and battery charger sizing are performed in accordance with IEEE Std 485 (Reference 8.3.2-2) and IEEE Std 946 (Reference 8.3.2-1), respectively.

Main dc power system equipment ratings are shown in Table 8.3.2-3.

8.3.2.3.4 Equipment Protection and Coordination Studies

The dc power equipment protection and coordination are performed in a manner similar to the ac power system described in Subsection 8.3.1.3.4.

8.3.2.3.5 Power Quality Limits

Onsite power quality limits are described in Subsection 8.3.1.3.~~5~~6 which includes considering harmonic contribution from the dc power system (i.e. battery chargers).

Table 8.3.1-1 Electrical Equipment Ratings - Component Data (Sheet 2 of 3)

Main ac Power System (Nominal Values)

4.	Generator Load Break Switch (GLBS) Rated Voltage Rated Current Rated Frequency	 Over 27.3kV Over 44.4kA 60Hz
5.	Isolated Phase Busduct (IPB) – Main Circuit Type Rated voltage Rated current Rated frequency	 Forced air cooling (Cooling air is cooled by water) Over ~~27.3~~28kV Over 44.4kA 60Hz
6.	Isolated Phase Busduct (IPB) – Branch Circuit Type Rated voltage Rated current Rated frequency	 Forced air cooling (Cooling air is cooled by water) Over ~~27.3~~28kV Over ~~5600A~~5900A 60Hz
7.	13.8kV Medium Voltage System	Non-Class 1E
	Switchgear Type Rated current	N1 & N2 Metal Clad 3000A
	Circuit Breaker Maximum voltage Rated short-circuit current Peak current (C & L crest) Control power	 15kV 50kA 130kA 125V dc

Table 8.3.1-3 Electrical Load Distribution - UAT/RAT Loading (Sheet 1 of 3)

Normal Operation

Load	Rated Output of Load [kW]	Load Factor [%]	Efficiency [%]	Power Factor [%]	Input of Load			Quantity Installed	Quantity Operating	UAT 1 RAT 1		UAT 2 RAT 2		UAT 3 RAT 3										UAT 4 RAT 4									
										13.8kV winding		13.8kV winding		6.9kV winding										6.9kV winding									
										N1 Bus		N2 Bus		A Bus		B Bus		N3 Bus		N4 Bus		P1 Bus		C Bus		D Bus		N5 Bus		N6 Bus		P2 Bus	
					[kW]	[kVAR]	[kVA]			Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]
Reactor Coolant Pump	6000	100	95	85	6316	3915	7431	4	4									1	7431	1	7431							1	7431	1	7431		
[[Circulating Water Pump	2300	95	90	65	2428	2839	3736	8	8	4	14944	4	14944																				]]
Motor Driven Main Feed Water Pump	11000	95	90	85	11612	7198	13662	4	4	2	27324	2	27324																				
Condensate Pump	3500	95	90	85	3695	2292	4348	3	2*											1	4348							1	4348	1	4348		
[[Secondary System Cooling Tower Fan	13500	95	90	85	14250	8832	16765	2	2	1	16765	1	16765																				]]
[[Makeup Water Pump	2300	95	90	85	2428	1506	2857	2	1*											1	2857									1	2857		]]
Turbine Component Cooling Water Pump	560	95	90	85	592	368	697	3	2*									1	697			1	697									1	697
Low Pressure Feed Water Heater Drain Pump	500	95	90	85	528	329	622	3	3									1	622	1	622									1	622		
Auxiliary Building Exhaust Fan	375	95	90	85	396	246	466	3	2*									1	466	1	466							1	466				
[[Customer Equipment	3200kVA	80	100	100	2560	0	2560	2	2	1	2560	1	2560																				]]
Insulator Washing Pump	840	50	90	85	467	291	550	2	1*											1	550									1	550		
Emergency Feed Water Pump	590	73	90	85	475	295	559	2	0																								
Safety Injection Pump	900	95	90	85	950	589	1118	4	0																								
Essential Service Water Pump	720	95	90	85	760	473	895	4	2*					1	895	1	895							1	895	1	895						
Component Cooling Water Pump	610	95	90	85	644	400	758	4	2*					1	758	1	758							1	758	1	758						
Containment Spray/Residual Heat Removal Pump	400	95	90	85	423	263	498	4	0																								
Charging Pump	820	95	90	85	866	537	1019	2	1*					1	1019											1	1019						
Control Rod Drive Mechanism Cooling Fan	315	95	90	85	333	207	392	2	1*													1	392									1	392
Non-Essential Chiller Unit	450	95	90	85	475	295	559	4	3*													2	1118									2	1118
[[Blowdown Pump	380	95	90	85	402	249	473	2	1*									1	473									1	473				]]
A Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1					1	2000																		
B Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1							1	2000																
C Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1															1	2000								
D Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1																	1	2000						
P1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1													1	2250										
P2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																							1	2250
N1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1	1	2250																						
N2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1			1	2250																				
N3 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1									1	2250														
N4 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1											1	2250												
N5 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																			1	2250				
N6 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																					1	2250		
[[Total Bus Capacity [kVA]											63843		63843		4672		3653		11939		18524		4457		3653		4672		14968		18058		4457]]
[[Transformer Capacity [kVA]											63843		63843										43245										45808]]

*: The quantity that is necessary for operation of a plant is this number. This number does not match up to a total of the quantity of operation of mentions in a column of each transformer.

Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.

Table 8.3.1-3 Electrical Load Distribution - UAT/RAT Loading (Sheet 2 of 3)
Start-up/Shutdown

Load	Rated Output of Load [kW]	Load Factor [%]	Efficiency [%]	Power Factor [%]	Input of Load			Quantity Installed	Quantity Operating	UAT 1 RAT 1 13.8kV winding		UAT 2 RAT 2 13.8kV winding		UAT 3 RAT 3 6.9kV winding										UAT 4 RAT 4 6.9kV winding									
										N1 Bus		N2 Bus		A Bus		B Bus		N3 Bus		N4 Bus		P1 Bus		C Bus		D Bus		N5 Bus		N6 Bus		P2 Bus	
					[kW]	[kVAR]	[kVA]			Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]
Reactor Coolant Pump	7800	100	95	85	8211	5089	9660	4	4									1	9660	1	9660							1	9660	1	9660		
[[Circulating Water Pump	2300	95	90	65	2428	2839	3736	8	8	4	14944	4	14944																				]]
Motor Driven Main Feed Water Pump	11000	95	90	85	11612	7198	13662	4	4	2	27324	2	27324																				
Condensate Pump	3500	95	90	85	3695	2292	4348	3	1*											1	4348									1	4348		
[[Secondary System Cooling Tower Fan	13500	95	90	85	14250	8832	16765	2	2	1	16765	1	16765																				]]
[[Makeup Water Pump	2300	95	90	85	2428	1506	2857	2	1*											1	2857									1	2857		]]
Turbine Component Cooling Water Pump	560	95	90	85	592	368	697	3	2*									1	697			1	697									1	697
Low Pressure Feed Water Heater Drain Pump	500	95	90	85	528	329	622	3	3									1	622	1	622									1	622		
Auxiliary Building Exhaust Fan	375	95	90	85	396	246	466	3	2*									1	466	1	466							1	466				
[[Customer Equipment	3200kVA	80	100	100	2560	0	2560	2	2	1	2560	1	2560																				]]
Insulator Washing Pump	840	50	90	85	467	291	550	2	1*											1	550									1	550		
Emergency Feed Water Pump	590	73	90	85	475	295	559	2	0																								
Safety Injection Pump	900	95	90	85	950	589	1118	4	0																								
Essential Service Water Pump	720	95	90	85	760	473	895	4	4					1	895	1	895							1	895	1	895						
Component Cooling Water Pump	610	95	90	85	644	400	758	4	4					1	758	1	758							1	758	1	758						
Containment Spray/Residual Heat Removal Pump	400	95	90	85	423	263	498	4	0																								
Charging Pump	820	95	90	85	866	537	1019	2	2					1	1019											1	1019						
Control Rod Drive Mechanism Cooling Fan	315	95	90	85	333	207	392	2	1*													1	392									1	392
Non-Essential Chiller Unit	450	95	90	85	475	295	559	4	3*													2	1118									2	1118
[[Blowdown Pump	380	95	90	85	402	249	473	2	1*									1	473									1	473				]]
A Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1					1	2000																		
B Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1							1	2000																
C Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1															1	2000								
D Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1																	1	2000						
P1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1													1	2250										
P2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																							1	2250
N1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1	1	2250																						
N2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1			1	2250																				
N3 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1									1	2250														
N4 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1											1	2250												
N5 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																			1	2250				
N6 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																					1	2250		
[[Total Bus Capacity [kVA]											63843		63843		4672		3653		14168		20753		4457		3653		4672		12849		20287		4457]]
[[Transformer Capacity [kVA]											63843		63843										47703										45918]]

*: The quantity that is necessary for operation of a plant is this number. This number does not match up to a total of the quantity of operation of mentions in a column of each transformer.

Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.

Table 8.3.1-3 Electrical Load Distribution - UAT/RAT Loading (Sheet 3 of 3)
Steam Generator Tube Rupture

Load	Rated Output of Load [kW]	Load Factor [%]	Efficiency [%]	Power Factor [%]	Input of Load			Quantity Installed	Quantity Operating	UAT 1 RAT 1		UAT 2 RAT 2		UAT 3 RAT 3										UAT 4 RAT 4									
										13.8kV winding		13.8kV winding		6.9kV winding										6.9kV winding									
										N1 Bus		N2 Bus		A Bus		B Bus		N3 Bus		N4 Bus		P1 Bus		C Bus		D Bus		N5 Bus		N6 Bus		P2 Bus	
					[kW]	[kVAR]	[kVA]			Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]	Q	[kVA]
Reactor Coolant Pump	6000	100	95	85	6316	3915	7431	4	3*									1	7431	1	7431							1	7431	1	7431		
[[Circulating Water Pump	2300	95	90	65	2428	2839	3736	8	8	4	14944	4	14944																				]]
Motor Driven Main Feed Water Pump	11000	95	90	85	11612	7198	13662	4	0																								
Condensate Pump	3500	95	90	85	3695	2292	4348	3	2*											1	4348							1	4348	1	4348		
[[Secondary System Cooling Tower Fan	13500	95	90	85	~~41250~~ 14250	8832	16765	2	2	1	16765	1	16765																				]]
[[Makeup Water Pump	2300	95	90	85	2428	1506	2857	2	1*											1	2857									1	2857		]]
Turbine Component Cooling Water Pump	560	95	90	85	592	368	697	3	2*									1	697			1	697									1	697
Low Pressure Feed Water Heater Drain Pump	500	95	90	85	528	329	622	3	0																								
Auxiliary Building Exhaust Fan	375	95	90	85	396	246	466	3	2*									1	466	1	466							1	466				
[[Customer Equipment	3200kVA	80	100	100	2560	0	2560	2	2	1	2560	1	2560																				]]
Insulator Washing Pump	840	50	90	85	467	291	550	2	1*											1	550									1	550		
Emergency Feed Water Pump	590	73	90	85	475	295	559	2	2							1	559							1	559								
Safety Injection Pump	900	95	90	85	950	589	1118	4	4					1	1118	1	1118							1	1118	1	1118						
Essential Service Water Pump	720	95	90	85	760	473	895	4	4					1	895	1	895							1	895	1	895						
Component Cooling Water Pump	610	95	90	85	644	400	758	4	4					1	758	1	758							1	758	1	758						
Containment Spray/Residual Heat Removal Pump	400	95	90	85	423	263	498	4	4					1	498	1	498							1	498	1	498						
Charging Pump	820	95	90	85	866	537	1019	2	0																								
Control Rod Drive Mechanism Cooling Fan	315	95	90	85	333	207	392	2	0																								
Non-Essential Chiller Unit	450	95	90	85	475	295	559	4	2													1	559									1	559
[[Blowdown Pump	380	95	90	85	402	249	473	2	1*									1	473									1	473				]]
A Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1					1	2000																		
B Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1							1	2000																
C Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1															1	2000								
D Station Service Transformer	2500kVA	80	-	85	1700	1054	2000	1	1																	1	2000						
P1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1													1	2250										
P2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																							1	2250
N1 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1	1	2250																						
N2 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1			1	2250																				
N3 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1									1	2250														
N4 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1											1	2250												
N5 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																			1	2250				
N6 Station Service Transformer	2500kVA	90	-	85	1913	1185	2250	1	1																					1	2250		
[[Total Bus Capacity [kVA]											36519		36519		5828		5269		11317		17902		3506		5828		5269		14968		17436		3506]]
[[Transformer Capacity [kVA]											36519		36519										43822										47007]]

*: The quantity that is necessary for operation of a plant is this number. This number does not match up to a total of the quantity of operation of mentions in a column of each transformer.

Note: The horsepower and equipment ratings are preliminary and typical, and are subject to change during detailed design.

Table 8.3.1-9 Electrical Equipment Location List (Sheet 4 of 7)

Name	Location		
	Bldg/F	Elevation	Room
A-Class 1E Battery	PS/B B1F	EL -26'-4"	A-Class 1E Battery Room
B-Class 1E Battery	PS/B B1F	EL -26'-4"	B-Class 1E Battery Room
C-Class 1E Battery	PS/B B1F	EL -26'-4"	C-Class 1E Battery Room
D-Class 1E Battery	PS/B B1F	EL -26'-4"	D-Class 1E Battery Room
Presserizer Heater Distribution Panel 1	R/B 4F	EL 76'-5"	CRDM Cabinet Room
Presserizer Heater Distribution Panel 2	R/B 4F	EL 76'-5"	CRDM Cabinet Room
P11-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
P21-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
~~N11~~N31-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
~~N21~~N41-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
~~N31~~N51-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
~~N41~~N61-Non-Class 1E Motor Control Center	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N11-UPS Unit	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N11-Non-Class 1E AC120V Switch Board	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N12-UPS Unit	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N12-Non-Class 1E AC120V Switch Board	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N21-UPS Unit	T/B ~~2F~~1F	EL 3'-7" ~~34'-0"~~	Electrical Room
N21-Non-Class 1E AC120V Switch Board	T/B ~~2F~~1F	EL 3'-7" ~~34'-0"~~	Electrical Room
N22-UPS Unit	T/B 2F	EL 34'-0" ~~3'-7"~~	Electrical Room
N22-Non-Class 1E AC120V Switch Board	T/B 2F	EL 34'-0" ~~3'-7"~~	Electrical Room
N31-UPS Unit	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room
N31-Non-Class 1E AC120V Switch Board	A/B 2F	EL 28'-6"	Non-Class 1E Electrical Room

Table 8.3.1-9 Electrical Equipment Location List (Sheet 6 of 7)

Name	Location		
	Bldg/F	Elevation	Room
P1-Non-Class 1E 480V Load Center	T/B 2F	EL 34'-0"	Electrical Room
P2-Non-Class 1E 480V Load Center	T/B 1F	EL 3'-7"	Electrical Room
N1-Non-Class 1E 480V Load Center	T/B 2F	EL 34'-0"	Electrical Room
N2-Non-Class 1E 480V Load Center	T/B 1F	EL 3'-7"	Electrical Room
N3-Non-Class 1E 480V Load Center	T/B 2F	EL 34'-0"	Electrical Room
N4-Non-Class 1E 480V Load Center	T/B ~~1F~~2F	EL 34'-0" ~~3'-7"~~	Electrical Room
N5-Non-Class 1E 480V Load Center	T/B 1F	EL 3'-7"	Electrical Room
N6-Non-Class 1E 480V Load Center	T/B 1F	EL 3'-7"	Electrical Room
P12-Non-Class 1E Motor Control Center	T/B ~~2F~~1F	EL 3'-7" ~~34'-0"~~	Turbine Building Local ~~Electrical Room~~
P22-Non-Class 1E Motor Control Center	T/B 1F	EL 3'-7"	Turbine Building Local ~~Electrical Room~~
N1-Non-Class 1E Motor Control Center	T/B ~~2F~~3F	EL 61'-0" ~~34'-0"~~	Turbine Building Local ~~Electrical Room~~
N2-Non-Class 1E Motor Control Center	T/B ~~1F~~3F	EL 61'-0" ~~3'-7"~~	Turbine Building Local ~~Electrical Room~~
N32-Non-Class 1E Motor Control Center	T/B ~~2F~~3F	EL 61'-0" ~~34'-0"~~	Turbine Building Local ~~Electrical Room~~
N42-Non-Class 1E Motor Control Center	T/B ~~2F~~1F	EL3'-7" ~~34'-0"~~	Turbine Building Local ~~Electrical Room~~
N52-Non-Class 1E Motor Control Center	T/B ~~1F~~3F	EL 61'-0" ~~3'-7"~~	Turbine Building Local ~~Electrical Room~~
N62-Non-Class 1E Motor Control Center	T/B 1F	EL 3'-7"	Turbine Building Local ~~Electrical Room~~
N3-Non-Class 1E Battery Charger	T/B 2F	EL 34'-0"	Electrical Room
N3-Non-Class 1E DC125V Switch Board	T/B 2F	EL 34'-0"	Electrical Room
N4-Non-Class 1E Battery Charger	T/B 1F	EL 3'-7"	Electrical Room
N4-Non-Class 1E DC125V Switch Board	T/B 1F	EL 3'-7"	Electrical Room
N34-Non-Class 1E Battery Charger	T/B 2F	EL 34'-0"	Electrical Room
N3-Non-Class 1E Battery	T/B 2F	EL 34'-0"	Electrical Room
N4-Non-Class 1E Battery	T/B 1F	EL 3'-7"	Electrical Room

Table 8.3.1-10 Motor Operated Valve List

Valves	Quantity	Current
Charging line isolation valve	1	15 A
RCP seal water return line isolation valve	1	7 A
Emergency feed water pump actuation valve	~~1~~2	~~25~~50 A
Hot leg sampling line CV isolation valve	2	7 A
Refueling water storage line isolation valve	1	15 A
Instrument air line isolation valve	1	7 A
RCP cooling water line isolation valve	4	45 A
Containment fan cooler unit cooling water line isolation valve	1	45 A
Total		~~308~~333 A

Note : This table shows list for one train.

Table 8.3.2-1 125V DC Class 1E Load Current Requirement
(Sheet 1 of 4)
Train A

Load Description	Normal Current (A)	Maximum Load Current		
		0 to 1 min (A)	1 to 119 min (A)	119 to 120 min (A)
A Switchboard Control Circuit	2	2	2	2
A Class 1E 6.9kV Switchgear	4	44	4	34
A Class 1E 480V Load Center	4	24	4	4
A Class 1E GTG Control Board	1	5	5	5
A Class 1E GTG Exciter	1	175	0	0
A UPS Unit	438	438	438	438
A&B MOV Inverter	1	1440	1	1
A Reactor Building DC Distribution Panel	11	15	11	11
A Solenoid Valve Distribution Panel	20	20	20	20
A Battery Charger Control Circuit	2	2	2	2
A Emergency Lighting	10	10	10	10
A MCR Radiation Monitor Pump	30	30	30	30
Total	524	2205	527	557
Random Load			For One Minute - 195	195

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-1 125V DC Class 1E Load Current Requirement
(Sheet 2 of 4)
Train B

Load Description	Normal Current (A)	Maximum Load Current		
		0 to 1 min (A)	1 to 119 min (A)	119 to 120 min (A)
B Switchboard Control Circuit	2	2	2	2
B Class 1E 6.9kV Switchgear	4	44	4	34
B Class 1E 480V Load Center	4	24	4	4
B Class 1E GTG Control Board	1	5	5	5
B Class 1E GTG Exciter	1	175	0	0
B UPS Unit	438	438	438	438
A&B MOV Inverter	1	1440	1	1
B Reactor Building DC Distribution Panel	11	15	11	11
B Solenoid Valve Distribution Panel	20	20	20	20
B Battery Charger Control Circuit	2	2	2	2
B Emergency Lighting	10	10	10	10
A MCR Radiation Monitor Pump	30	30	30	30
Total	524	2205	527	557
Random Load			For One Minute - 195	195

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-1 125V DC Class 1E Load Current Requirement

(Sheet 3 of 4)

Train C

Load Description	Normal Current (A)	Maximum Load Current		
		0 to 1 min (A)	1 to 119 min (A)	119 to 120 min (A)
C Switchboard Control Circuit	2	2	2	2
C Class 1E 6.9kV Switchgear	4	44	4	34
C Class 1E 480V Load Center	4	24	4	4
C Class 1E GTG Control Board	1	5	5	5
C Class 1E GTG Exciter	1	175	0	0
C UPS Unit	438	438	438	438
C&D MOV Inverter	1	1440	1	1
C Reactor Building DC Distribution Panel	11	15	11	11
C Solenoid Valve Distribution Panel	20	20	20	20
C Battery Charger Control Circuit	2	2	2	2
C Emergency Lighting	10	10	10	10
B MCR Radiation Monitor Pump	30	30	30	30
Total	524	2205	527	557
Random Load			For One Minute - 195	195

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-1 125V DC Class 1E Load Current Requirement

(Sheet 4 of 4)

Train D

Load Description	Normal Current (A)	Maximum Load Current		
		0 to 1 min (A)	1 to 119 min (A)	119 to 120 min (A)
D Switchboard Control Circuit	2	2	2	2
D Class 1E 6.9kV Switchgear	4	44	4	34
D Class 1E 480V Load Center	4	24	4	4
D Class 1E GTG Control Board	1	5	5	5
D Class 1E GTG Exciter	1	175	0	0
D UPS Unit	438	438	438	438
C&D MOV Inverter	1	1440	1	1
D Reactor Building DC Distribution Panel	11	15	11	11
D Solenoid Valve Distribution Panel	20	20	20	20
D Battery Charger Control Circuit	2	2	2	2
D Emergency Lighting	10	10	10	10
B MCR Radiation Monitor Pump	30	30	30	30
Total	524	2205	527	557
Random Load			For One Minute - 195	195

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-2 125V DC Non-Class 1E Load Current Requirement
(Sheet 1 of 4)
N1 System

Load Description	Normal Current (A)	Maximum load Current			
		0 to 1 min (A)	1 to 5 min (A)	5 to 30 min (A)	30 to 60 min (A)
N1 Switchboard Control Circuit	2	2	2	2	2
Non-Class 1E UPS Unit N12	0	525	525	525	0
Non-Class 1E UPS Unit N32	0	525	525	525	0
Non-Class 1E UPS Unit N42	0	525	525	0	0
A/B Distribution Panel N1	30	30	30	30	30
A/B Distribution Panel N2	40	40	40	40	40
Solenoid Valve Distribution Panel	6	6	6	6	6
MCR Radiation Monitor Pump	30	30	30	30	30
Total	108	1683	1683	1158	108

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-2 125V DC Non-Class 1E Load Current Requirement
(Sheet 2 of 4)
N2 System

Load Description	Normal Current (A)	Maximum load Current			
		0 to 1 min (A)	1 to 5 min (A)	5 to 30 min (A)	30 to 60 min (A)
N2 Switchboard Control Circuit	2	2	2	2	2
Non-Class 1E UPS Unit N11	0	525	525	525	0
Non-Class 1E UPS Unit N31	0	525	525	525	0
Non-Class 1E UPS Unit N41	0	525	525	0	0
Non-Class 1E UPS Unit N5	0	525	525	0	0
A/B Distribution Panel N3	30	30	30	30	30
A/B Distribution Panel N4	24	24	24	24	24
Main Control Room Distribution Panel	30	30	30	30	30
Solenoid Valve Distribution Panel	9	9	9	9	9
Total	95	2195	2195	1145	95

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

Table 8.3.2-2 125V DC Non-Class1E Load Current Requirement
(Sheet 3 of 4)
N3 System

Load Description	Normal Current (A)	Maximum load Current			
		0 to 1 min (A)	1 to 30 min (A)	30 to 31 min (A)	31 to 60 min (A)
N3 Switchboard Control Circuit	2	2	2	2	2
Vacuum Breaker	3	3	3	183	3
Main Generator Seal pump	1	450	350	350	350
T/B Distribution Panel A	30	30	30	30	30
Transformer Auxiliary System Distribution panel	30	30	30	30	30
Non-Class 1E 6.9kV P1 Switchgear	4	44	4	4	4
Non-Class 1E 480V P1 Load Center	5	27	5	5	5
Non-Class 1E 13.8kV N1 Switchgear	4	60	4	4	4
Non-Class 1E 6.9kV N3 Switchgear	4	44	4	4	4
Non-Class 1E 6.9kV N4 Switchgear	4	44	4	4	4
Non-Class 1E 480V N1 Load Center	5	27	5	5	5
Non-Class 1E 480V N3 Load Center	5	27	5	5	5
Non-Class 1E 480V N4 Load Center	5	27	5	5	5
Non-Class 1E UPS Unit N22	0	525	525	0	0
Secondary System Solenoid Valve Distribution Panel	9	9	9	9	9
Total	111	1349	985	640	460

Note: The DC loads are preliminary and typical, and are subject to change during detailed design.

undervoltage relays.

2. Load requirement for auxiliary parts
   Multifunction relay: 0.15A
   Auxiliary relay: 0.1A
   Indication light: 0.1A
   Undervoltage relay: 0.05A

3. The DC loads are preliminary and typical, and are subject to change during detailed design without impact to the DCD.

Table 8.3.2-3 Electrical Equipment Ratings - Component Data (Sheet 1 of 2)

Class 1E DC Power System

(Nominal Values)

a. Battery Bank 4 - 125Vdc, 2 strings of 60 flooded lead acid cells, 5000Ah*, float voltage 2.25V/cell, equalize voltage 2.33V/cell, 8 hr rating
b. Battery Charger 4 - ac input – 480V, 3 phase, 60Hz; dc output – 125Vdc, 700A Continuous; float voltage – 135V, equalizing charge voltage 140V, 24hr recharge
c. Switchboard 4 – 125Vdc, Main bus 3000A, 50kA short circuit
d. Panelboards 4 - Main bus 225A continuous, 40kA short circuit
e. Spare Battery Charger (Non-class 1E) 2 - ac input – 480V, 3 phase, 60Hz; dc output – 125Vdc, 700A Continuous; float voltage – 135V, equalizing charge voltage 140V, 24hr recharge

*: Actual Ah rating is decided in accordance with manufacturer's specification.

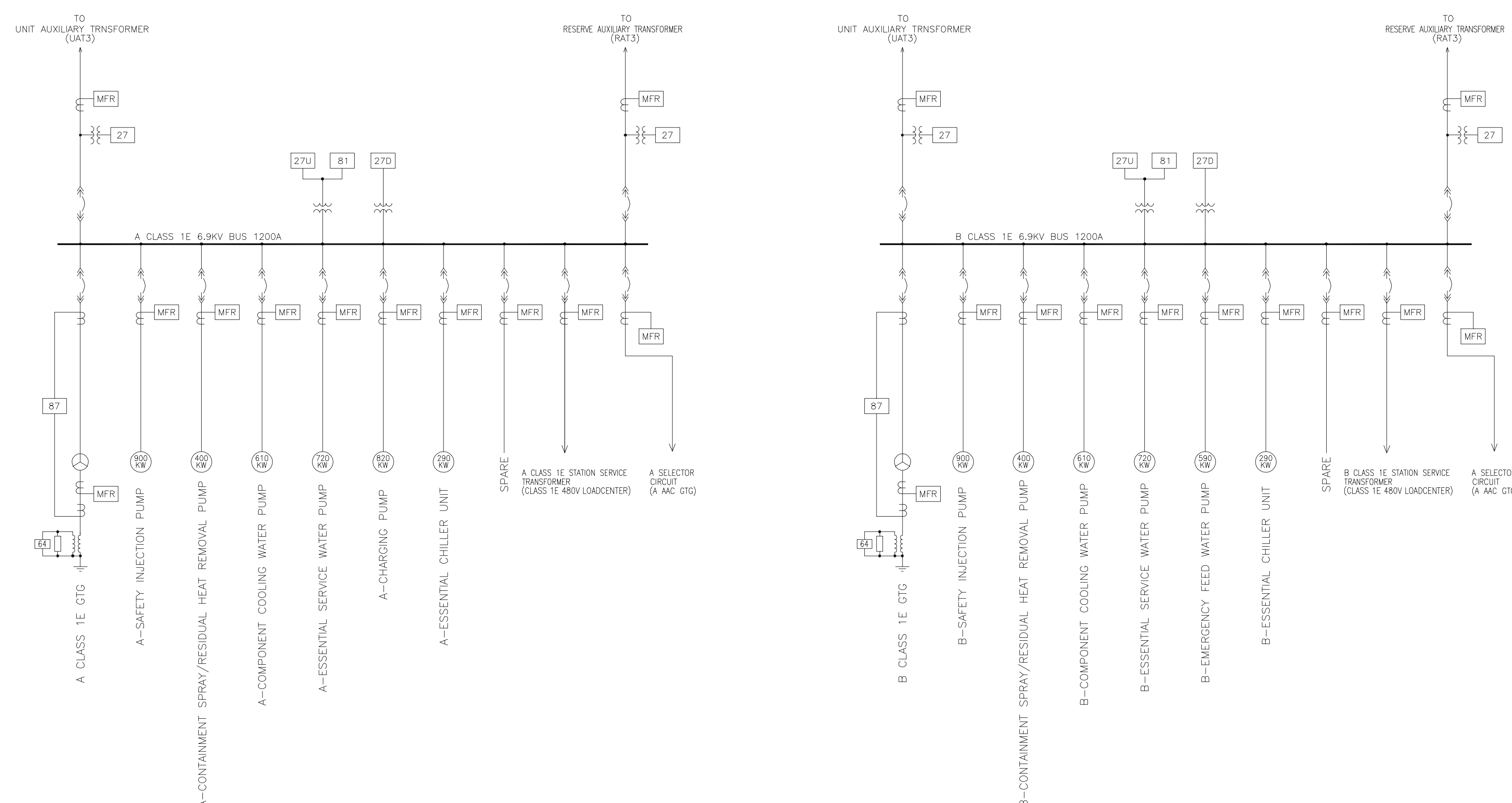


Figure 8.3.1-1 Onsite AC electrical distribution system (Sheet 2 of 7)

Class 1E 6.9kV buses A and B one line diagram

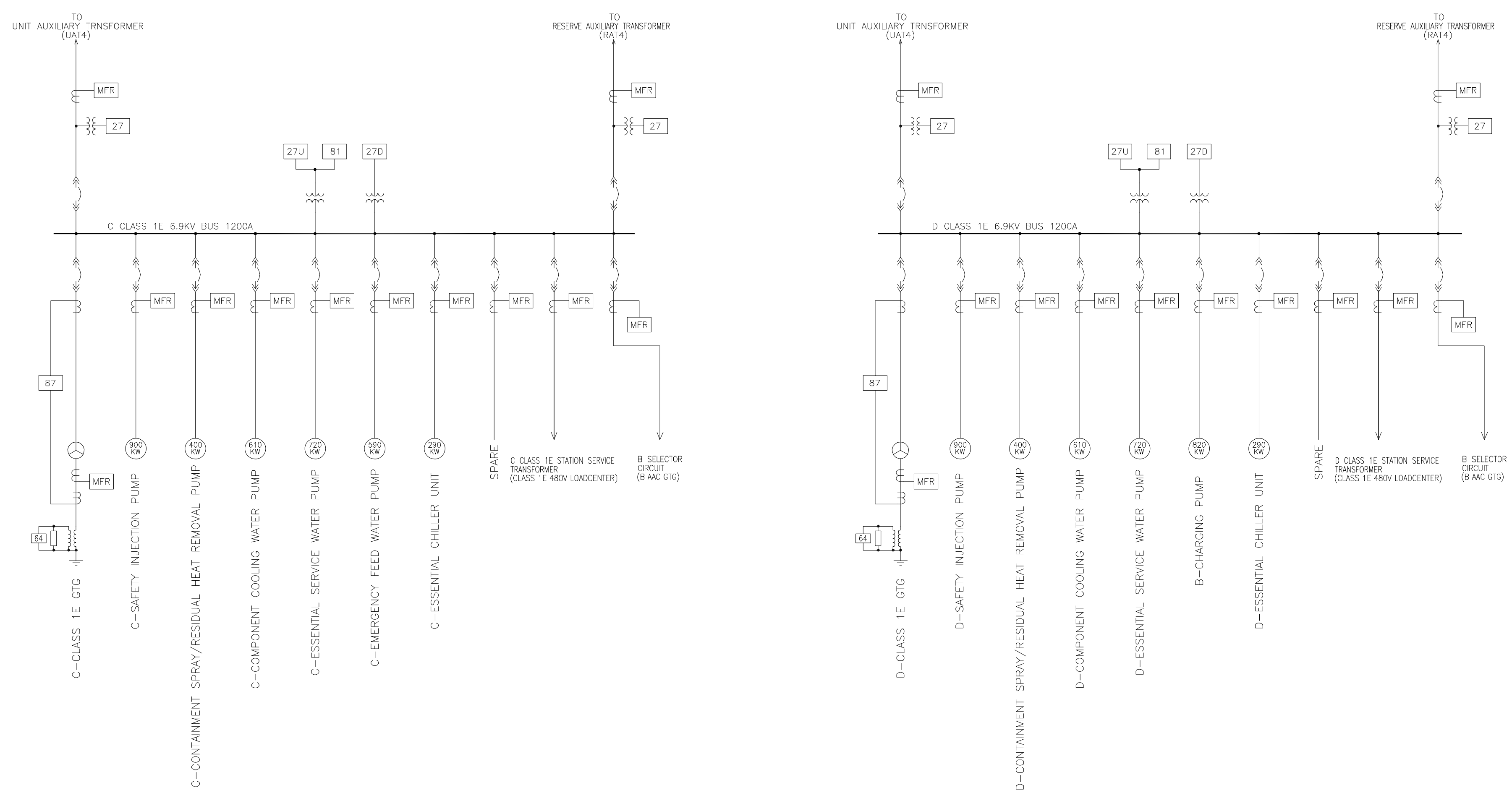


Figure 8.3.1-1 Onsite AC electrical distribution system (Sheet 3 of 7)

Class 1E 6.9kV buses C and D one line diagram

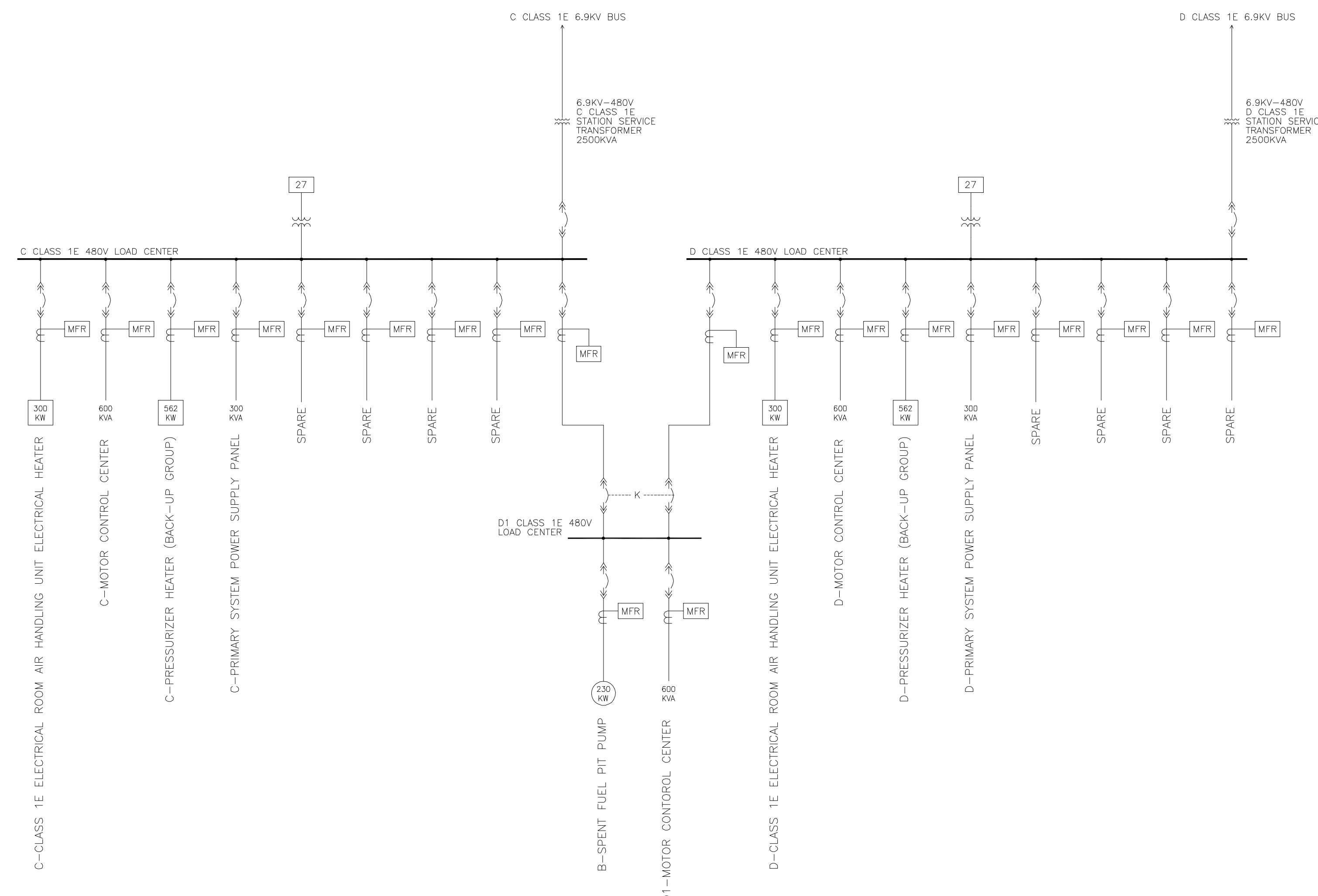


Figure 8.3.1-1 Onsite AC electrical distribution system (Sheet 6 of 7)

Class 1E 480V buses C and D one line diagram

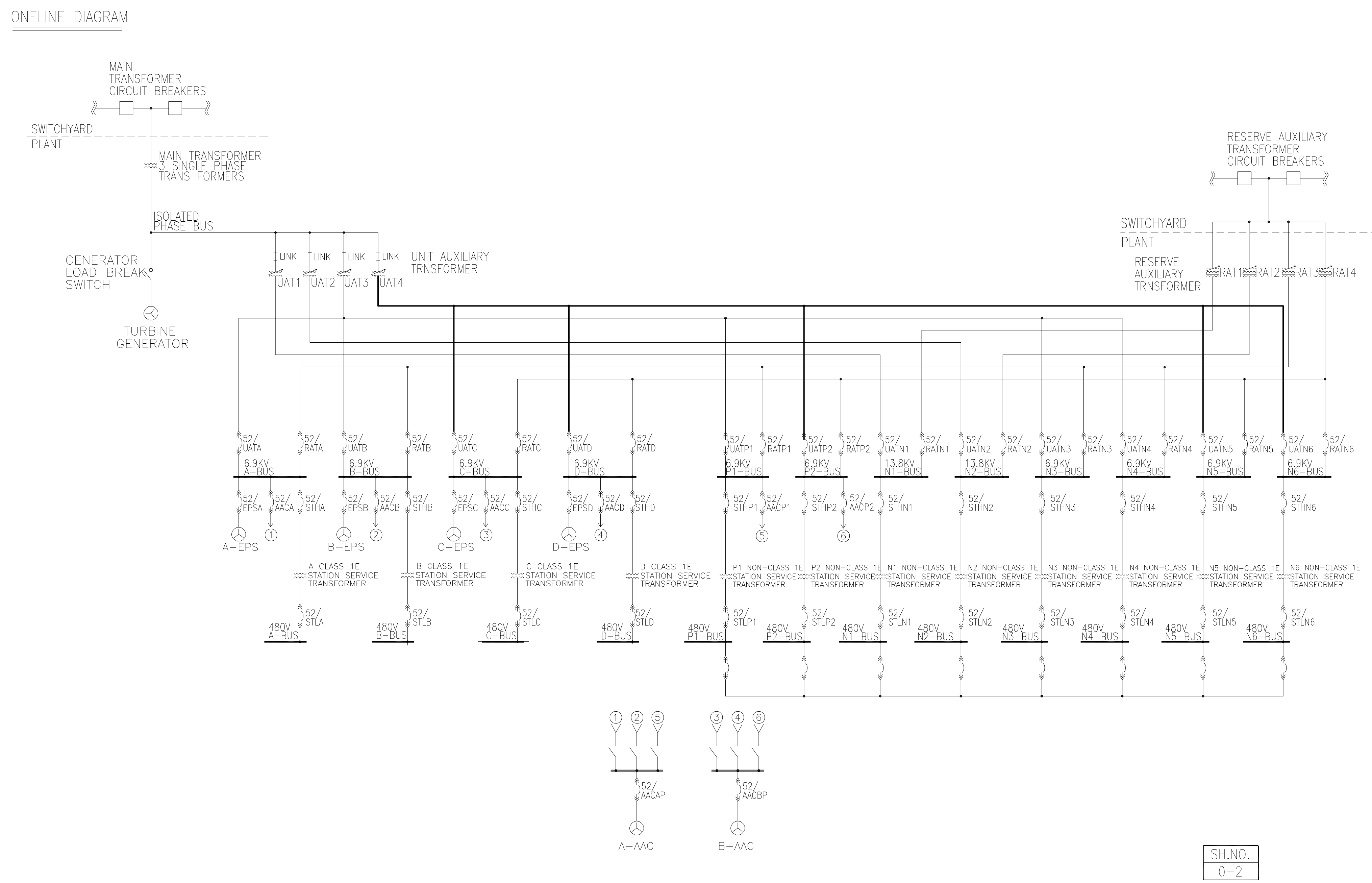


Figure 8.3.1-2 Logic diagrams (Sheet 2 of 24)

One line diagram

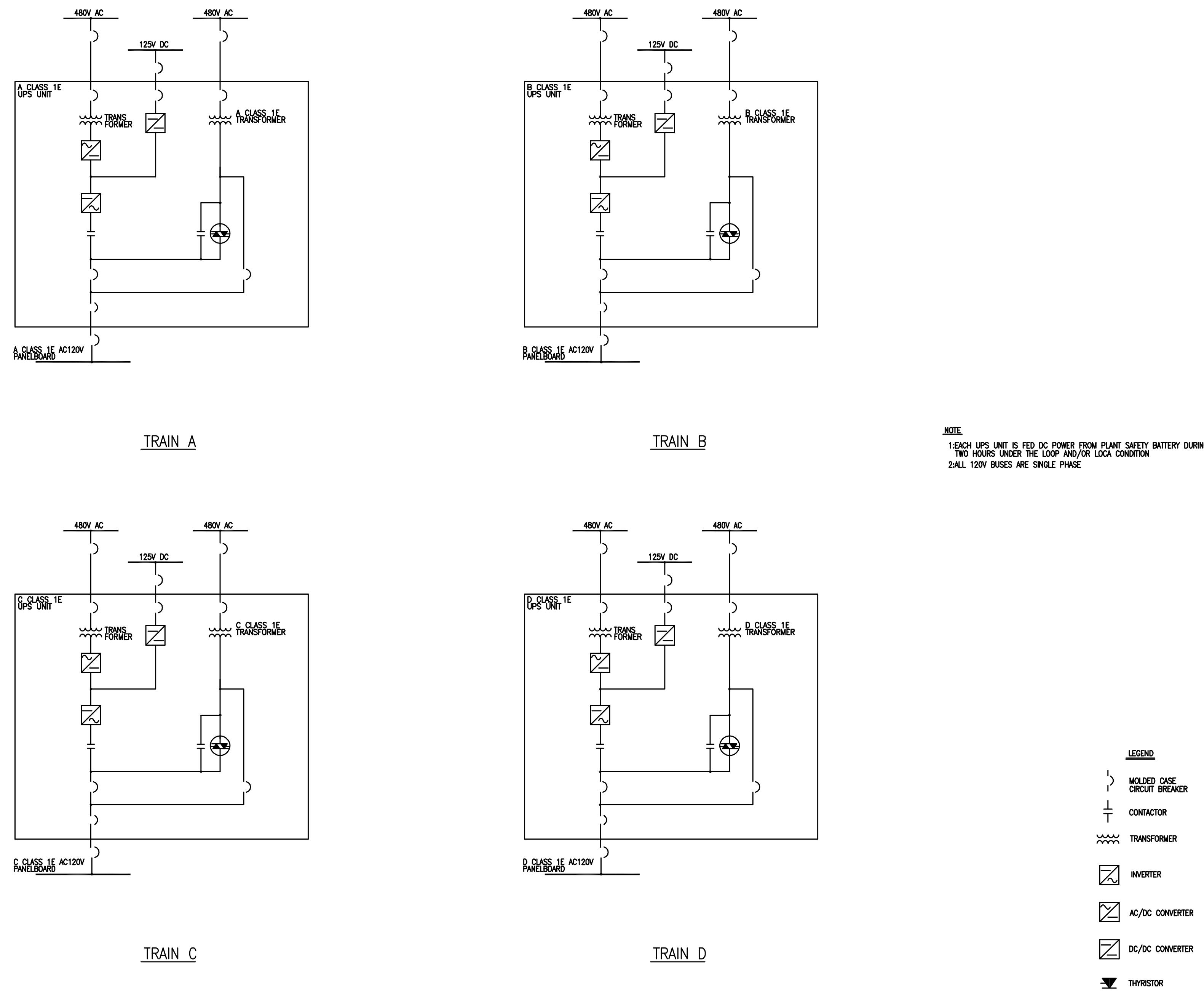


Figure 8.3.1-3 120V AC I&C power supply panels (Sheet 1 of 2)

Class 1E

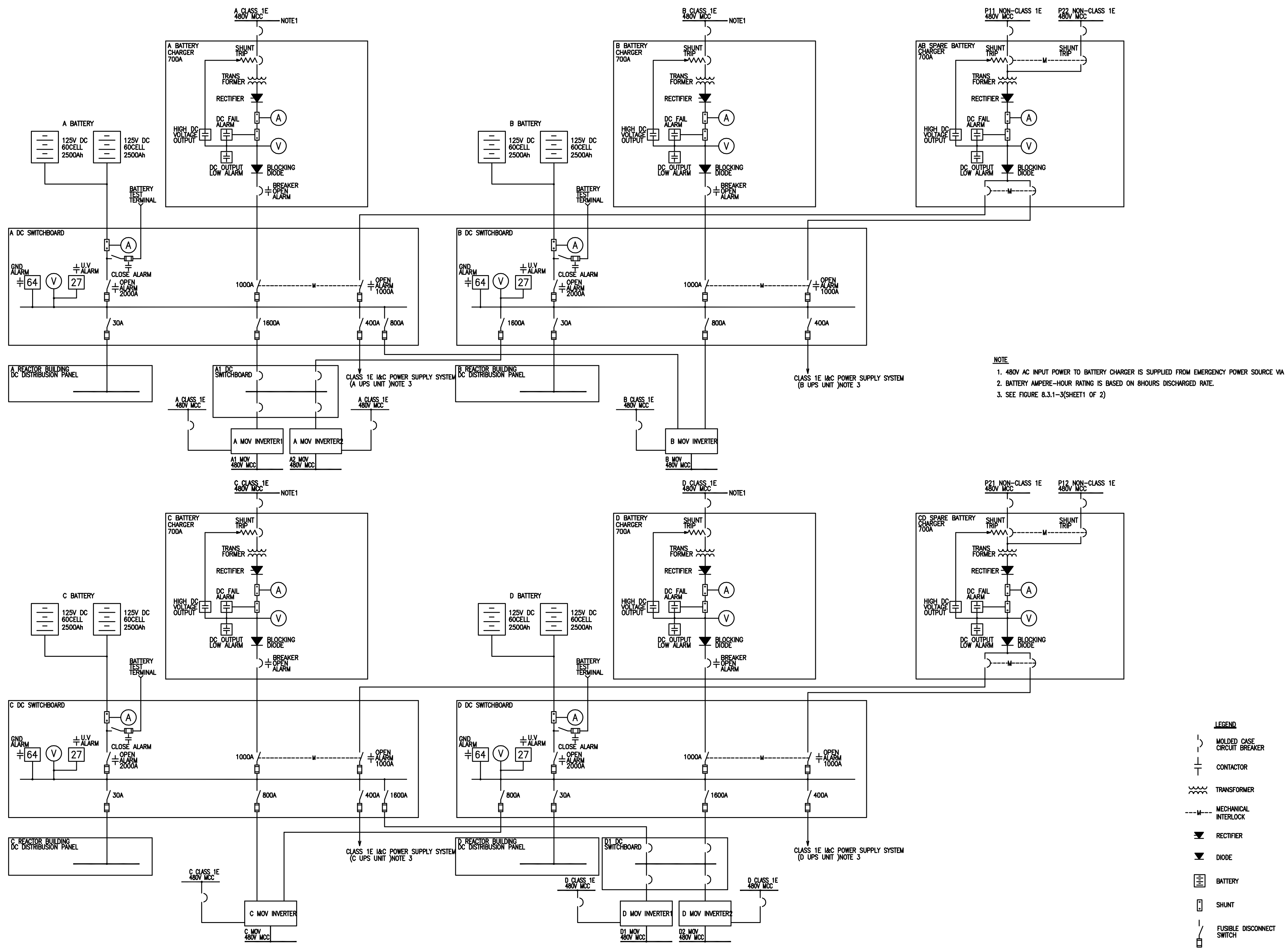


**Figure 8.3.2-1 DC Power Distribution System (Sheet 1 of 2)
Class 1E**

8.4.1.3 Alternate AC Power Sources

AAC power sources and their connections to the onsite and offsite ac power systems meet the requirements of RG 1.155 (Reference 8.3.1-21).

Two full capacity 4000 kW, 6.9 kV non-Class 1E GTGs (A and B) are provided as AAC sources and any one of these two GTGs can meet the SBO load requirements shown in Table 8.3.1-6 for the time required to bring and maintain the plant in a safe shutdown condition. Two AAC GTGs are provided for operational flexibility and enhanced reliability, even though the provision of one AAC GTG is adequate to meet the requirements of RG 1.155 (Reference 8.3.1-21). RG 1.155 Appendix B (Reference 8.3.1-21) does not require a single failure criterion to be applicable to the AAC power source. Hence, the provision of two 100% capacity AAC sources will provide greater US-APWR reliability for coping with an SBO event than what is intended by RG 1.155 (Reference 8.3.1-21). The AAC power sources reach set voltage and frequency within 100 seconds from receiving the starting signal. Controls exist in the MCR to start, stop and synchronize the AAC power sources.

To minimize the potential for common mode failures with the Class 1E GTGs, different rating GTGs with diverse starting system are provided as AAC sources. The auxiliary and support systems for the AAC GTGs are independent and separate from the Class 1E GTGs to minimize the potential for common mode failures. Completely separate and independent fuel supply systems and onsite fuel storage tanks are provided for the Class 1E GTGs and for the non-Class 1E AAC GTGs.

The A-AAC GTG and B-AAC GTG are located in separate rooms in the PS/B. A-AAC GTG is connected to the non-Class 1E 6.9 kV permanent bus P1 through a selector circuit A. Similarly, B-AAC GTG is connected to the non-Class 1E 6.9 kV permanent bus P2 through a selector circuit B. The selector circuit consists of one circuit breaker connected to the AAC source and three disconnect switches. The disconnect switches in the selector circuit A are connected to the 6.9 kV buses P1, A and B (or P2, C and D for selector circuit B) through tie lines, as shown in Figure 8.3.1-1.

The A-AAC GTG and B-AAC GTG are connected to the circuit breakers in selector circuits A and B, respectively. The selector circuits A and B are located in the PS/B. The non-Class 1E 6.9 kV and 480 V permanent power supply systems P1 and P2 are located in the T/B electrical room. These AAC GTG circuit breakers in the selector circuits A and B are normally open and the AAC power sources are not normally connected directly to the plant offsite or onsite power system. The Class 1E circuit breakers in the Class 1E MV switchgears are connected to the disconnect switches (non-Class 1E) in the selector circuits A and B as shown in Figure 8.3.1-1. The non-Class 1E disconnect switches in the selector circuits A and B, and the Class 1E incoming circuit breakers in the Class 1E MV switchgear from the AAC GTG are normally open and do not have any automatic closing function. They perform the isolation between the Class 1E and the non-Class 1E system. This meets RG 1.155, Appendix B (Reference 8.3.1-21) requirements for isolation between AAC sources and the onsite and offsite power systems.

The different rating and diverse starting mechanism of the AAC sources from the emergency ac power sources, the location of AAC sources in separate rooms, the

Chapter 9

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9-xi	Acronyms and Abbreviations	Acronym was reviewed due to the adjustment with the System code
9.1-17	9.1.3.2.1.3	R-COLA RAI 4206 (CP RAI #135) Question 12.03-12.04-11 Added the operating experience of SFP heat exchanger discussed in EPRI TR 1013470 to minimize leakage from plate-type heat exchanger.
9.1.26	9.1.4.2.1.1	Acronym was reviewed due to the adjustment with the System code
9.1-30	9.1.4.2.2.2	Acronym was reviewed due to the adjustment with the System code
9.1-33 to 34	9.1.4.4 9.1.4.5	Acronym was reviewed due to the adjustment with the System code
9.1-37	9.1.5.1	Correction (editorial corrections) Replace “applicant” to “Applicant”
9.1-38	9.1.5.2.2	Corrected the limitation of the movement for spent fuel cask handling crane and spent fueling handling crane from “rail stop” to “electrical inter lock”.
9.1-46	9.1.7	R-COLA RAI 4206 (CP RAI #135) Question 12.03-12.04-11 Added “Plant Support Engineering: Guidance for Replacing Heat Exchangers at Nuclear Power Plants with Plate Heat Exchangers, July 2006” as the reference including operating experience of SFP heat exchanger.
9.1-53	Table 9.1.5-1 Table 9.1.5-2	Correction (editorial corrections) Replace Power “460V” to “480V” Replace Space Heater “230V” to “120V” Replace “ATSM” to “ASTM” Replace polar crane rail top level from “Elevation 145’-6” “ to ”Elevation 145’-7” “. Replace capacity of polar crane from “250 ton” to “270 ton”. Replace traveling speed of polar crane bridge from “0.9, 1.8, 18.0” to “0.6, 3.42, 12.0” and trolley from “0.6, 3.42, 12.0” to “0.9, 1.8, 18.0”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.1-55	Table 9.1.5-4	Replace polar crane maximum load rating from “250” to “270”.
9.2-4, 6 to 8, 11,14	9.2.1.2.1 9.2.1.2.2.1 9.2.1.2.2.5 9.2.1.2.3.1 9.2.1.3	Correction (editorial corrections) Replace “applicant” to “Applicant”
9.2-18	9.2.2.1.2.2	Correction (editorial corrections) Replace “The CCWSis” with “The CCWS is”.
9.2-19	9.2.2.2 5th paragraph	Correction (editorial corrections) Replace “is branch off” with “branches”.
9.2-19	9.2.2.2 7th paragraph	Correction (editorial corrections) Replace “the system. The CCW water the filters to protect the plate type CCW heat exchangers are not deemed necessary and not provided.” with “the system, therefore, the CCW filter is not necessary”.
9.2-20	9.2.2.2.1.2 5th paragraph	Correction (editorial corrections) Replace “Since the difference of installation elevation between the surge tanks and the pumps is large enough, as NPSH available, there is sufficient margin.” with “The surge tanks are located at a higher elevation than the pumps to ensure sufficient NPSH margin is available.”.
9.2-21	9.2.2.2.1.3 9th paragraph	Correction (editorial corrections) Replace “in the surge tank as a countermeasure of the negative pressure in a tank at the time of a sudden fall of tank” with “on the surge tank to prevent damaging the tank in the event of a sudden decrease in”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-21	9.2.2.2.1.5 First bullet 2nd paragraph	Correction (editorial corrections) Replace "Header isolation can be attained even if assumming single failure, since there are two header tie line isolation valves. Since a header tie line isolation valve will be closed in about 10 seconds or less, it is satisfactory to isolate by S+UV signal, P signal, and surge tank water low-low level." with "Header isolation meets the single failure criteria by incorporating two header tie line isolation valves. The header isolation valves are designed to close within 30 seconds upon a S+UV signal, P signal, or surge tank water low-low level.".
9.2-21-22	9.2.2.2.1.5 First bullet 2nd paragraph	Correction (editorial corrections) Replace "the valve close signal currently sent is made to bypass and the valve is made to open. CCW pumps are designed such that one CCW pump can supply water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) during normal operation. Therefore, the header isolation valves are maintained to be open." with "the isolation signal can be bypassed and the isolation valves respond. In addition, the header isolation valves are opened in order to supply cooling water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) by one CCW pump during normal operation.".

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-22	9.2.2.2.1.5 7th bullet First paragraph	Correction (editorial corrections) Replace “Two air-operated isolation valves are provided in series on each CCW supply line isolation valves are provided on each CCW supply line (A2 and C2) to the components located in the non-seismic category I buildings (turbine building (T/B) and auxiliary building (A/B). These valves close to protect against CCW seismic category I out-leakage through the non-seismic category I portions by automatic closure upon the demand signals” with “The CCW system supplies cooling water to components located in the non-seismic Category I buildings (turbine building and auxiliary building). Each CCW supply line (A2 and C2) has two in-series air operated isolation valves. These valves close automatically to isolate the non-seismic Category I portion of the CCW system upon receipt of a S+UV signal, P signal or surge tank low-low level signal.”.
9.2-23	9.2.2.2.1.5 7th bullet 2nd paragraph	Correction (editorial corrections) Replace “CCW out-leakage through the non-seismic CCW return lines (A2 and C2) is prevented by check valves series located on the return line for components lecated in the non-seismic Category I buildings (i.e. the turbine (T/B) and auxiliary building (A/B))” with “In-series check valves are provided on the CCW return lines from the non-seismic Category I portion of the CCW system”.
9.2-23	9.2.2.2.1.5 8th bullet	Correction (editorial corrections) Deleted “These Valves function to supply cooling water to the RCPs of header A-1 (or C-1) in the event cooling is lost due to a single failure during on-line maintenance of a CCW pump. The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-24	9.2.2.2.1.5 12th bullet	Correction (editorial corrections) Added "The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).".
9.2-25	9.2.2.2.2.4 First paragraph	Correction (editorial corrections) Replace "The signal to the pump is setting up delay time." with "The start signal to the pumps is delayed.".
9.2-25	9.2.2.2.2.6 First paragraph	Correction (editorial corrections) Replace "The CCWS is designed in consideration of the water hammer prevention and mitigation of its in accordance with the following as discussed in NUREG-0927." with "The CCWS is designed in consideration of water hammer prevention and mitigation in accordance with the following as discussed in NUREG-0927.".
9.2-29 to 30	9.2.4.1	Correction (editorial corrections) Replace "applicant" to "Applicant" Add "The" before "COL Applicant"
9.2-32	9.2.4.2.3	Correction (editorial corrections) Add "The" before "COL Applicant"
9.2-34	9.2.5.1	Correction (editorial corrections) Replace "applicant" to "Applicant"
9.2-40	9.2.5.3 9.2.5.4	Correction (editorial corrections) Replace "applicant" to "Applicant"
9.2-59	9.2.9.2.1	Reflected the additional comment of COL applicant
9.2-63	9.2.10	Correction (editorial corrections) COL 9.2(11) and 9.2(13) are redundant. COL 9.2(11) is replaced with COL 9.2(13) and COL 9.2(13) is deleted.
9.2-63	9.2.10	Correction (editorial corrections) COL 9.2(14) and 9.2(16) are redundant. COL 9.2(16) is deleted.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-63	9.2.10 COL 9.2(9),(10) and (12)	Correction (editorial corrections) Add “The” before “COL Applicant”
9.2-64 to 65	9.2.10 COL 9.2(25),(26),(29),(31) and (32)	Correction (editorial corrections) Replace “applicant” to “Applicant”
9.2-84	Table 9.2.2-5	Design progress The flow rate of A2 train at normal power operation is changed to “1988gpm” from “1948gpm” as per vendor information.
9.2-84	Table 9.2.2-5	Design progress The total flow rate of A, B, A1 and A2 train at normal power operation is changed to “7163gpm” from “7123gpm” as per vendor information.
9.2-84	Table 9.2.2-5	Design progress The flow rate of A2 train at cooldown by CS/RHRS is changed to “1988gpm” from “1948gpm” as per vendor information.
9.2-84	Table 9.2.2-5	Design progress The flow rate of subtotal (A, B, A1 and A2 train) at cooldown by CS/RHRS is changed to “15963gpm” from “15923gpm” as per vendor information.
9.2-124	Figure 9.2.9-1	Correction (editorial corrections) Correct arrow mark for CWS.
9.3-14-15	9.3.2.2.5 4th paragraph	Design progress Change of the sentence about SG blow-down water quality monitoring system because the design shall be changed based on EPRI guideline
9.3-23	9.3.4.1.1	“safety injection” was replaced with “ECCS actuation” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.3-38	9.3.4.2.7.4	Correction (typographical correction) Delete redundant “in”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.3-39	9.3.4.3	"safety injection" was replaced with "ECCS actuation" Sentences "Therefore, the provision for a leakage detection and control program in accordance with 10 CFR 50.34 (f) (xxvi) does not apply." were deleted Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.3-44	9.3.4.5.3.7 1st paragraph	Correction (typographical correction) Replace "indicatef" with "indicate"
9.3-54	Table 9.3.1-2	Design progress The outlet dew point of IAS is changed from "-40°F" to "-58°F" to be consistent with ISA standard and EPRI Utility Requirements Document (URD).
9.3-60	Table 9.3.2-5	Design progress Change of the SG blow-down water quality monitoring system based on EPRI guideline (Change of a pH monitoring to SC, CC monitoring, and an addition of Na analysis monitoring. An addition of SG blow-down sample line in a secondary-system sampling station).
9.3-68	Table 9.3.4-3 (Sheet 2 of 6)	Design progress Change the design parameters from the value based on safety analysis conditions to the value based on operating conditions.
9.4-V	Acronyms and Abbreviations	Acronym was reviewed due to the adjustment with the System code

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.4-1	9.4 9.4.1	“emergency” was replaced with “abnormal” in section 9.4. The sentences “including accident condition and LOOP condition,” and “The Main Control Room Heating, Ventilation and Air Conditioning System is subjected to the design objectives of RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning” as it contains airborne radioactive material. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref.9.4.8-27) are summarized in Table 12.3-8. RG 4.21 is also applicable to the Auxiliary Building Ventilation System and the Engineered Safety Feature Ventilation System” were added to section 9.4. “and 4.21” was added to section 9.4.1 Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.4-14	9.4.3.1.2.4	Reflected the additional comment of COL applicant
9.4-23	9.4.3.5.4	“The TSC HVAC System is operated from MCR” was added. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.4-23	9.4.4	Correction (editorial correction) Replace "maintains" with "maintain”
9.4-24	9.4.4.1.2	Reflected the additional comment of COL applicant
9.4-24	9.4.4.1.2	Correction (editorial correction) Replace “design to maintains” with “designed to maintain”
9.4-24	9.4.4.2.1 1st paragraph	Correction (editorial correction) Replace “fan” with “fans”
9.4-25	9.4.4.2.1 3rd paragraph	Correction (Engineering progress) Replace “all 27 fans” with “all fans” to delete detail information.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.4-25	9.4.4.2.1 4th paragraph	Correction (editorial correction) Replace “fan” with “fans” Replace “a local thermostat and temperature controller” with “ local thermostats and temperature controllers” Replace “the area temperature” with “the area temperatures” Replace “A basement “ with “Basement” Replace “exhaust/circulating fan is” with “ exhaust fans are”
9.4-25	9.4.4.2.1 5th paragraph	Correction (Engineering progress) Replace “is be” with “is ” Replace “split unit type” with “the” to delete detail information.
9.4-25	9.4.4.2.2 2nd paragraph	Correction (editorial correction) Replace “serves to the” with “ serves the” Replace “in” with “into” Replace “Non-Class 1E” with “a Non-Class 1E” Replace “permanent bus” with “a permanent bus”
9.4-26	9.4.4.3	Correction (editorial correction) Replace “dose” with “does”
9.4-28 through 9.4-29	9.4.5.1.1.2 and 9.4.5.1.2	Correction Removed the following description of Subsection 9.4.5.1.1.2. “· Maintain the hydrogen concentration below 1% by volume of Class 1E battery room.” Added the following description to Subsection 9.4.5.1.2. “The Class 1E electrical room HVAC system is designed to maintain the hydrogen concentration below 1% by volume of Class 1E battery room.”

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.4-31	9.4.5.2.2	Design progress Replaced "Rooms with high heat loss during the cold season are provided with non safety-related unit heaters or in-duct electric heaters in their supply air branches. These electric heaters are classified as equipment class 5 and seismic category II." With "The safety-related in-duct heaters are provided in supply air branches to Remote Shutdown Console Room, Class 1E Battery Rooms, Class 1E I&C Rooms and Class 1E Electrical Room & MCR HVAC Equipment Rooms. These electric heaters are classified as equipment class 3 and seismic category I." due to changing of the safety classification for the Class 1E Electrical Room HVAC System In-duct heaters.
9.4-33	9.4.5.2.3	Sentence "and the air handling unit inlet damper and outlet damper open upon receipt of their respective fan run signals." was added. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.4-43	9.4.6.1.2.4 9.44.6.2.1	Reflected the additional comment of COL applicant
9.4-58	Table 9.4-1	"only" was added to Note:5 Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.4-63 through 9.4-64	Table 9.4.3-1	Correction Modified the equipment class of the following components to be consistent with Table 3.2-2 - Auxiliary Building Air Handling Unit - Auxiliary Building Exhaust Fan - Non-Class 1E Electrical Room Air Handling Unit - Non-Class 1E Electrical Room Return Air Fan - Main Steam / Feedwater Piping Areas Air Handling Unit - Technical Support Center Toilet/Kitchen Exhaust Fan

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.4-67	Table 9.4.4-1	Correction (Engineering progress) Delete air flow capacity and fan type for the air handling unit, numbers and flows for fans, to delete detail information
9.4-68	Table 9.4.5-1	Design progress Modified the cooling coil capacity of Class 1E Electrical Room Air Handling Unit (Train C, D) from 2,250,000 btu/hr to 2,290,000 btu/hr.
9.4-69	Table 9.4.5-1	RAI 670-4773, Question 09.04.05-17 Modified the cooling coil capacity of Emergency Feedwater Pump (T/D) Area Air Handling Unit from 60,000 btu/hr to 62,000 btu/hr.
9.4-72 through 9.4-78	Table 9.4.5-2	Design progress Added the failure modes and effects analysis of Class 1E Electrical Room HVAC System In-duct heater due to changing of the safety classification for the duct heaters.
9.4-88	Figure 9.4.4-1 sheet 1 of 2	Correction (editorial correction) Replace “supply fan” with “supply fans” for the basement area fans
9.4-90	Figure 9.4.5-1	Correction Revised Figure 9.4.5-1.
9.4-91	Figure 9.4.5-2	Design progress Revised Figure 9.4.5-2 due to changing of the safety classification for the Class 1E Electrical Room HVAC System In-duct heaters.
9.4-92	Figure 9.4.5-3	Correction Revised Figure 9.4.5-3.
9.5-x	Acronyms and Abbreviations	Acronym was reviewed due to the adjustment with the System code
9.5-1	9.5.1	Correction (typographical correction) Modified “dose” to ”does”.
9.5-13	9.5.1.2.6	”Fire detectors are to be provided for areas containing safety related equipment and initiate fire alarms.” was added. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.5-16	9.5.1.3	"shall provide a milestone for completing" and "FSAR" were added to Section 9.5.1.3. Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
9.5-16	9.5.1.5	"if a fire water storage tank is used," and "The fire pumps are operable from the MCR." were added to section 9.5.1.5. Reason: Reflected the additional comment of COL applicant
9.5-16	9.5.2	Correction (typographical correction) Modified "applicant" to "Applicant"
9.5-16	9.5.1.5	Correction Replace "The fire water storage tank is monitored for level and temperature. The diesel-driven fire pump fuel storage tank, if a diesel driven fire pump is used, is monitored for level." With "The fire water storage tank, if a fire water storage tank is used, is monitored for level and temperature. The diesel-driven fire pump fuel storage tank, if a diesel driven fire pump is used, is monitored for level.".
9.5-22	9.5.2.2.2	Correction (typographical correction) Modified "applicant" to "Applicant"
9.5-22	9.5.2.2.2.2 1st paragraph 5th bullet	Correction Replace "Offsite support center" with "Offsite emergency operations facility (EOF)"
9.5-22	9.5.2.2.2.2 2nd paragraph 1st sentence	Correction Replace with "Communication between the onsite technical center (TSC) and main control room (MCR) may be made using the PABX, station radio system, plant page system and the sound powered telephone system. The sound powered telephone system is an on site system and cannot be used to communicate with offsite facilities."
9.5-25	9.5.2.2.5.1 9.5.2.2.5.2	Correction (typographical correction) Modified "applicant" to "Applicant"

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.5-35	9.5.4.2.2.1	Correction (typographical correction) Modified “applicant” to “Applicant”
9.5-38	9.5.4.3	Correction (typographical correction) Modified “applicant” to “Applicant”
9.5-49 to 50	9.5.9	Reflected the additional comment of COL applicant Correction (typographical correction) Modified “applicant” to “Applicant”
9.5-95	Table 9.5.1-1 Sheet 40 of 46	Correction (typographical correction) Modified “applicant” to “Applicant”

ACRONYMS AND ABBREVIATIONS

A/B	auxiliary building
ac	alternating current
AC/B	access building
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BTP	branch technical position
CCWS	component cooling water system
CFR	Code of Federal Regulations
CGS	compressed gas system
COL	Combined License
CRDM	control rod drive mechanism
CRE	control room envelope
CS/RHRS	containment spray/residual heat removal system
C/V	containment vessel
CVCS	chemical and volume control system
CWS	circulating water system
dc	direct current
DCD	Design Control Document
DWS	demineralized water system
ECCS	emergency core cooling system
EIA	Energy Information Administration
EPRI	Electric Power Research Institute
ESF	engineered safety features
ESW	essential service water
ESWS	Essential Service Water System
FCC	Federal Communications Commission
FMEA	failure mode and effects analysis
FOS	fuel oil storage and transfer system
FSAR	Final Safety Analysis Report
~~FTS~~	~~Fuel Transfer System~~
GDC	General Design Criteria
GTG	gas turbine generator
GWMS	gaseous waste management system

The cooling and purification flow paths are shown in Figure 9.1.3-1 and Figure 9.1.3-2, respectively.

The purification portion of the SFPCS, i.e., piping, demineralizers, and filters, are non-safety related.

The equipment classification for the SFPCS is provided in Chapter 3, Section 3.2.

9.1.3.2.1 Component Description

The SFPCS component design parameters are provided in Table 9.1.3-3.

9.1.3.2.1.1 Spent Fuel Pit

The SFP is described in Subsection 9.1.2.

9.1.3.2.1.2 Spent Fuel Pit Pumps

Two identical pumps are installed in parallel in the SFPCS. Each pump is sized to circulate the pit water through the SFP heat exchanger in conjunction with the demineralizer and the filter to perform purification and cooling of the SFP.

The SFP pumps are horizontal centrifugal type, and the wetted area in contact with the fuel pit water is of stainless steel material.

9.1.3.2.1.3 Spent Fuel Pit Heat Exchangers

Two SFP heat exchangers are provided to remove decay heat from the SFP, as specified in Subsection 9.1.3.2.2.2. These heat exchangers are plate-type heat exchangers constructed of austenitic stainless steel. The SFP water circulates through one side of the heat exchanger while the CCW circulates through the other side. The design of SFP heat exchangers shall incorporate specific features regarding industry operating experience as discussed in EPRI TR 1013470 to minimize leakage from Plate type heat exchangers (Ref. 9.1.7-27).

9.1.3.2.1.4 Spent Fuel Pit Filters

Two vertical, cylindrical cartridge-type SFP filters are provided in the purification portion of the SFPCS. Each cartridge filter is designed for a flow rate of approximately 265 gpm. The filter is used to improve the pit water clarity by removing solid particles. The filter assembly is constructed of austenitic stainless steel with disposable filter cartridges.

9.1.3.2.1.5 Spent Fuel Pit Demineralizers

Two vertical, cylindrical demineralizers are provided, and each demineralizer is designed for a flow rate of approximately 265 gpm. The demineralizer removes ionic impurities from the SFP water before being circulated back to the SFP. The vessels are constructed of austenitic stainless steel.

The refueling machine transport fuel assemblies between the fuel transfer system ~~(FTS)~~ and the reactor core within the confines of the refueling cavity. The refueling machine consists of a bridge with two motorized end trucks which traverse the length of the refueling cavity. Mounted atop the bridge, is a vertical mast tube assembly which traverses the bridge perpendicular to the direction of the motorized end trucks. This provides an arrangement wherein the mast can be precisely indexed over a fuel assembly in the reactor core. The mast tube assembly contains a gripper mechanism which is lowered to latch onto a fuel assembly. The fuel assembly is then raised into the mast tube to protect the fuel assembly during transport. The mast tube also contains a sipping system used to detect leaking fuel.

The refueling machine also has an auxiliary hoist which is used in the control rod drive shaft unlatching operation.

Electrical interlocks, limit switches, and mechanical stops are utilized to prevent damage to a fuel assembly to assure appropriate radiation shielding depth below the water level in the refueling cavity, and to monitor the fuel assembly load for imparted loads greater than the nominal weight of the fuel assembly. Imparted loads could result from unidentified movement restrictions such as binding of the fuel assembly in the core.

9.1.4.2.1.2 Fuel Handling Machine

The fuel handling machine transport fuel assemblies between the fuel elevator and the SFP within the confines of the refueling area pits and fuel transfer canal. The fuel handling machine consists of a bridge with two motorized end trucks which traverse the length of the spent fuel pit, the cask pit and the fuel inspection pit. Mounted atop the bridge, is a vertical mast tube assembly which traverses the bridge perpendicular to the direction of the motorized end trucks. This provides an arrangement wherein the mast can be precisely indexed over a fuel assembly in the spent fuel rack. The mast tube assembly contains a gripper mechanism which is lowered to latch onto a fuel assembly which is then raised into the mast tube to protect the fuel assembly during transport.

The fuel handling machine also has an auxiliary hoist which is provided to handle the inserts for a new or a spent fuel assembly using appropriate handling tool. The auxiliary hoist also handles the gates separating the various pits (pools). The auxiliary hoist has the load capacity to lift a new or a spent fuel assembly using a spent fuel assembly handling tool, as backup the mast tube assembly. The auxiliary hoist has a load limiting device to prevent the hoist from exerting excessive force.~~ The auxiliary hoist has the load capacity to lift a fuel assembly, but is configured to preclude latching on to fuel assembly.~~

As for electrical interlock, limit switches, and mechanical stops, which is same function for refueling machine, are also provided for fuel handling machine.

9.1.4.2.1.3 Suspension Hoist on the Spent Fuel Cask Handling Crane

The suspension hoist on the spent fuel cask handling crane (Subsection 9.1.5) has a load limit interlock. This interlock precludes the suspension hoist from lifting a load greater than its rated capacity. In addition, administrative procedure defined in Subsection 13.5.1

- Verifying functionality of the reactor cavity lighting
- Verifying tools are in place and functional
- Verifying the ~~FTS~~ fuel transfer system is functional

After the reactor head bolting is de-tensioned, but prior to lifting the head and overflowing from the reactor vessel, the lower levels of the refueling cavity are flooded using a fill line which enters through the refueling cavity floor. This is done at flow rate which will minimize scattering of activated dust.

When the lower levels of the refueling cavity are flooded, the reactor vessel head assembly is unseated and raised 2.5 ft above the flange. At this point, disconnection of the control drive shafts is verified. Upon verification of disconnection, the reactor vessel head assembly is raised while maintaining a maximum of one foot clearance above the refueling cavity water to provide shielding.

When the water level reaches the normal refueling water level, the reactor vessel head assembly is transported to the lay down area. Concurrently, refueling cavity lighting and the refueling cavity water filtration system is placed in service.

The upper reactor internals with the in-core instrumentation system (ICIS) thimble assemblies is lifted using the lift rig with a load cell in the lift rigging. The load cell monitors the force applied when lifting the internals and provides indication of interference with other core structures and fuel assemblies. When the upper reactor internals is clear of the reactor vessel, it is transferred to its storage location in the lower refueling cavity. The core is then ready for refueling.

- Phase III – Fuel Handling

  All irradiated fuel assemblies are removed from the core and relocated to the SFP. The partially used fuel and new fuel assemblies are then transferred and installed into their designated positions in the reactor core.

  In general, the fuel handling procedure is as follows:

  - The refueling machine is indexed over a fuel assembly in the core.
  - The refueling machine mast latches onto a fuel assembly. The fuel assembly is raised to the designated height clearing the vessel flange while maintaining the established satisfactory radiation shielding depth below the water surface.
  - The fuel transfer car is moved into the containment from the fuel storage area where the fuel container is pivoted into the vertical position.
  - The refueling machine loaded with an irradiated fuel assembly traverses the reactor cavity until it is indexed over the vertical ~~FTS~~ fuel transfer system fuel container. The irradiated fuel assembly is lowered into the container and unlatched.

9.1.4.3 Safety Evaluation

The LLHS is evaluated as to its ability to assure there are no unacceptable releases of radiation as a result of mechanical damage to fuel, to prevent damage that could compromise the ability to maintain an adequate degree of subcriticality, to maintain acceptable shielding during fuel handling, withstand earthquakes, and to assure fuel handling is performed within acceptable limits.

- Damage to fuel assemblies is prevented by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13). This is further assured through the operating procedures defined in Subsection 13.5.2.

- Maintenance of subcriticality is achieved by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13).

- Maintenance of acceptable shielding requirements is achieved by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13). This is further assured through the operating procedures defined in Subsection 13.5.2.

- The ability to withstand natural phenomena, specifically earthquakes, is achieved by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13) using the seismic design criteria presented in Chapter 3.

- Fuel handling performance is assured to be within acceptable limits by designing and configuring the light load handling system to comply with ANS 57.1-1992 (Ref. 9.1.7-13). This is further assured through the operating procedures defined in Subsection 13.5.2.

9.1.4.4 Inspection and Testing Requirements

The inspection and testing requirements for the light load handling system are as outlined below:

- For the fuel handling machine, the new fuel elevator, the ~~FTS~~ fuel transfer system including upenders, and the refueling machine, the following shop tests are performed:

  - All hoists and cables are load tested to 125% of their rated load capacity

  - All equipment will be assembled and verified to conform to specified operational characteristics

- Prior to use, the following steps will be taken to assure the light load handling system is functional:

- Visual inspection for loose or foreign parts with maintenance to keep free of dirt and grease
- Lubrication of exposed gears with proper lubricant
- Inspection of hoist cables for worn or broken strands
- Visual inspection of all limit switches and limit switch actuators for any sign of damaged or broken parts
- Inspection and/or testing of the equipment for proper functional and running operation

• For fuel handling tools, the following shop tests are performed:

  - The tools are load tested to 125% of the rated load
  - The tools are assembled and checked for proper functional operation

• Prior to use, the following steps will be taken to assure the light load handling system is functional:

  - Visual inspection of the tools for dirt and loose hardware and for any signs of damage such as nicks and burrs
  - Check of tools for proper functional operation

9.1.4.5 Instrumentation Requirements

The light load handling system has a system of instrumentation and controls (interlocks), alarms, and communication devices to assure the light load handling system meets the criterion discussed in Subsection 9.1.4.1. The interlocks provided are as defined in ANS 57.1, paragraph 6.3.1.1, and in Table 1 for the fuel handling machine, the new fuel elevator, the ~~FTS~~fuel transfer system including upenders, and the refueling machine.

The light load handling system has interlock actuation annunciation lamps on the control console to visually prompt the operator of interlock status. Additionally, movement of the fuel handling machine and the refueling machine bridge are audibly signaled.

The plant is designed with a public address system. The fuel handling machine, the new fuel elevator, the ~~FTS~~fuel transfer system including up enders, and the refueling machine is to have the capability to be interlinked with the public address system in the fuel handling area and the PCCV at a minimum. Additionally, administrative procedure defined in Subsection 13.5.1 provides communication devices not susceptible to a loss of power, offsite, or onsite, such as sound powered telephones or two-way radios. These are to be used to provide communication between operators at the fuel handling machine, the new fuel elevator, the ~~FTS~~fuel transfer system including upenders, and the refueling machine. These devices operate on channels or frequencies unique to the light load non-critical heavy loads, as defined in Section 9.1.5 above, that, because of theirlocaiton, timing, and the load path could not cause a significant release of radioactivity, cause a loss of margin to criticality, uncover irradiated fuel in the reactor vessel or spent fuel pool, or damage equipment essential to achieve or maintain safe shutdown. Non-critical lifts would be evaluated and documented in a manner similar to a critical heavy load lift, as required by the heavy load handling program to be developed by the COL Applicant as required by COL 9.1 (6) and Subsection 9.1.5.3 of this DCD.

The areas of the plant in which the OHLHS is operated are shown in Figures 9.1.5-1 through 9.1.5-4. These figures represent the Fuel Handling Area and the interior of the PCCV. The OHLHS is designed to meet requirements of 10 CFR 50, Appendix A, specifically, GDC 1, 2, 4, and 5.

The operation, testing, maintenance, and inspection of OHLHS are controlled utilizing ~~through the use of~~ safe load paths as defined in Figures 9.1.5-1 through 9.1.5-4 and administrative control procedures.

The administrative control procedures govern the operation, testing, maintenance, and inspection of overhead heavy load handling system. These procedures incorporate the requirements of and follow the recommendations and/or guidelines of the following documents:

Scope	Reference	Reference Title
General requirements	Chapter 5, Section 5.1.1, NUREG-0612	Control of Heavy Loads at Nuclear Power Plants (Ref. 9.1.7-21)
Crane Operators (Training, qualifications, and conduct.)	Chapter 2-3, ANSI/ASME B30.2	Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist (Ref. 9.1.7-22)
Inspection, testing, and maintenance.	Chapter 2-2, ANSI/ASME B30.2	Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist (Ref. 9.1.7-22)

9.1.5.2 System Description

The primary pieces of equipment used in the OHLHS are the spent fuel cask handling crane in the fuel handling area, equipment hatch hoist in the PCCV and the polar crane in the PCCV. The spent fuel cask handling crane, equipment hatch hoist and the polar crane are designed in accordance with the provisions of NUREG-0554 and ASME NOG-1 as Type I single-failure-proof cranes. Therefore these cranes are designed to retain control of and continue to hold their maximum loads during a~~n~~ SSE. The OHLHS is seismic category II and Equipment Class 5, as described in Section 3.2.

Other than the single-failure-proof OHLHS, miscellaneous hoists and cranes with heavy load capacities are installed in safety-related areas of the US-APWR plant. Descriptions and data for all cranes and hoists that have heavy load capacities ~~and~~which are installed over safe shutdown equipment are given in Table 9.1.5-3. The safety evaluations for those cranes and hoists are discussed in Subsection 9.1.5.3.

The OHLHS also includes equipment accessories (e.g., slings, and hooks, etc.) instrumentation, physical stops and/or electrical interlocks, and associated administrative controls.

The applicable Codes and Standards are identified in Section 9.1.5.1.

9.1.5.2.1 Physical Arrangement

The areas of the plant in which the spent fuel cask handling crane and polar crane operate are shown in Figures 9.1.5-1 through 9.1.5-4. The specifications for the spent fuel cask handling crane and the polar crane are given in Table 9.1.5-1 and 9.1.5-2. As shown, the spent fuel handling crane has three load handling hooks, the main, the auxiliary, and the suspension crane. The suspension crane is only used for new fuel assembly handling between a new fuel container to the new fuel storage area or between the new fuel storage rack and the basket on the new fuel elevator. Because of this limitation, the suspension crane is considered part of the light load handling system. Its operation and control is detailed in Section 9.1.4.

9.1.5.2.2 Spent Fuel Cask Handling Crane

A spent fuel cask filled with spent fuel assemblies is lifted and transferred using the main hoist of the spent fuel cask handling crane and the spent fuel cask lift rig. The cask's path is from the cask loading pit to the truck access area on the ground floor as shown on Figure 9.1.5-1.

Neutron source containers and Irradiation sample containers are transferred using the auxiliary hoist through the path shown on Figure 9.1.5-2.

A reactor coolant pump (RCP) motor is transferred from the PCCV into the fuel handling area. In the fuel handling area, once the RCP motor is in position, it is lifted by the main ~~hook~~hoist of the spent fuel handling crane and transferred to the truck access area using the path shown on Figure 9.1.5-3.

Miscellaneous equipment is transferred from the PCCV using the same path as the RCP motors. The spent fuel cask handling crane movement and storage is handled as follows:

- The spent fuel handling cask crane range of movement is limited; in general, to the fuel handling area defined by the ~~hook~~hoist coverage ranges shown in Figure 9.1.5-1. The limitation is controlled by the configuration of the spent fuel handling cask crane and by electrical interlock ~~permanent rail stops installed on the crane rails~~.

- For the RCP motors and miscellaneous equipment, movement is design limited to exclude the new fuel storage, cask, and fuel inspection pits. The movement of the spent fuel handling crane is limited by electrical interlock ~~removable rail stops~~.

9.1.7-18 “Packaging and Transportation of Radioactive Material,” Energy. Title 10, Code of Federal Regulations, Part 71, U.S. Nuclear Regulatory Commission, Washington, DC.

9.1.7-19 Single-Failure-Proof Cranes for Nuclear Power Plants. NUREG-0554, U.S. Nuclear Regulatory Commission, Washington, DC, May 1979.

9.1.7-20 Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). ASME NOG-1, 2004, American Society of Mechanical Engineers.

9.1.7-21 Control of Heavy Loads at Nuclear Power Plants. NUREG-0612, U.S. Nuclear Regulatory Commission, Washington, DC, July 1980.

9.1.7-22 Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist). ANSI/ASME B30.2-2005, American Society of Mechanical Engineers.

9.1.7-23 American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials. American National Standards Institute, ANSI N14.6-1993, American Nuclear Society, IL.

9.1.7-24 Slings. ANSI/ASME B30.9-2003, American Society of Mechanical Engineers.

9.1.7-25 Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes. CMAA Specification No.70, 2000, Crane Manufacturers Association of America, Inc.

9.1.7-26 Thermal-Hydraulic Analysis for US-APWR Spent Fuel Racks, MUAP-09014P (R0) and MUAP-09014NP (R0), Mitsubishi Heavy Industries, Ltd.,June 2009.

9.1.7-27 Plant Support Engineering: Guidance for Replacing Heat Exchangers at Nuclear Power Plants with Plate Heat Exchangers, July 2006

Table 9.1.5-1 Specification of the Spent Fuel Cask Handling Crane

1. Type		Overhead bridge crane		
2. Operating device		Radio remote control unit and cab on crane		
3. Component supplied electric power		Trolley		
4. Electric power supply		Power	: ~~460~~480V ac, 60 Hz, 3 Phase	
		Space Heater	: ~~230~~120V ac, 60 Hz, Single Phase	
5. Bridge Span		47'-3"		
6. Top level of the rail		Elevation 125'-8"		
		Main ~~Hook~~Hoist	Auxiliary ~~Hook~~Hoist	Suspension Hoist
7. Capacity	Metric ton	150	20	2
8. Lift	ft-in (m)	124'-9" (38.003 m)	124'-9" (38.003 m)	69'-3" (21.0886 m)
9. ~~Hook~~Hoist Coverage	ft-in (m)	Refer to Figure 9.1.5-1 and 9.1.5-2		
10. Hoisting Speed	m/min	0.12, 0.6, 1.2	0.45, 1.8, 4.5	2.1, 6.3
11. Traveling Speed	m/min	Bridge: 0.6, 1.5, 6.0		Suspension Crane: 3.0, 9.0
		Trolley: 0.6, 1.5, 6.0		Hoist: 3.0, 9.0
12. Wire Material		Stainless Steel (ASTM~~ATSM~~ A 492 Type 304)		

Table 9.1.5-2 Specification of the Polar Crane

1. Type		Overhead bridge crane	
2. Operating device		Portable wireless control box on operating floor, Cab on crane	
3. Component supplied electric power		Trolley	
4. Electric power supply		Power : ~~460~~480V ac, 60 Hz, 3 Phase	
		Space Heater : ~~230~~120V ac, 60 Hz, Single Phase	
5. Bridge Span		142'-1"	
6. Top level of the rail		Elevation 145'-7~~6~~"	
		Main ~~Hook~~Hoist	Auxiliary ~~Hook~~Hoist
7. Capacity	Metric ton	~~250~~270	50
8. Lift	ft-in (m)	67'-9" (20.650 m)	119'-1" (36.296 m)
9. ~~Hook~~Hoist Coverage	ft-in (m)	Refer to Figure 9.1.5-4	
10. Hoisting Speed	m/min	0.12, 0.6, 1.2	1.2, 2.4~~6.0~~, 3.0~~12.0~~
11. Traveling Speed	m/min	Bridge: ~~0.9, 1.8, 18.0~~0.6,3.42,12.0	
		Trolley: ~~0.6, 3.42, 12.0~~0.9,1.8,18.0	
12. Wire Material		Carbon Steel	

Table 9.1.5-~~3~~4 Cranes and Hoists Installed Over Safe Shutdown Equipment

Crane and Hoist	Crane/Hoist Type		Location	Maximum Load Rating (metric tons)	ASME NOG-1 Type	Single-Failure-proof	Seismic Category
Polar Crane	Top-Running Overhead Bridge Crane	Main hoist	PCCV	~~250~~270	I	Yes	II
		Auxiliary hoist		50	I	Yes	
Spent Fuel Cask Handling Crane	Top-Running Overhead Bridge Crane	Main hoist	R/B(Fuel handling area)	150	I	Yes	II
		Auxiliary hoist		20	NA	No	
		Suspension hoist		2	NA	No	
MSIV(main steam isolation valve)room crane	Underhung overhead crane		R/B (MS/FW Piping Area hung from roof slab)	10	NA	No	II
PCCV Equipment Hatch Hoist	Base mounted Drum Hoist		PCCV (above equipment hatch at azimuth 40°)	40	~~NA~~I	~~No~~Yes	II
Safety Injection Pump(SIP) Room Hoist	Monorail Hoist		R/B(SIP Rooms, Floor EL.-26'-4")	5	NA	No	II
CS/RHR Pump Room Hoist	Monorail Hoist		R/B(CS/RHR Pump Rooms, Floor EL.-26'-4")	5	NA	No	II
EFW Pump Room Hoist	Monorail Hoist		R/B(EFW Pump Rooms, Floor EL.-26'-4")	5	NA	No	II
CCW Pump Hoist	Monorail Hoist		R/B(CCW Rooms, Floor EL.-26'-4")	5	NA	No	II
CCW Heat Exchanger Hoist	Monorail Hoist			2	NA	No	II
Essential Chiller Unit Hoist	Monorail Hoist		East and West PS/B(Basement Floor EL.-26'-4")	3	NA	No	II

component failure in one train coincident with on-line maintenance in another train do not prevent the ESWS from performing its safety-related functions with the two remaining operable trains. Instrumentation is also provided independently and not shared among the trains.

Each supply line after the strainer is tapped to supply cooling water to each component. Each CCW HX is provided with piping and isolation valves around the heat exchanger~~,~~ which facilitates back flushing of the CCW HX of the ESW side when required. Heat from the reactor auxiliaries is removed from the CCW HX and the heated service water flows to the UHS via independent lines. The ESW flow of 13,000 gpm is maintained at all operating conditions, including accident conditions and safe shutdown with a LOOP. The ESWS is designed to operate at s water temperature as low as 32° F. For the ESWS piping and components in the R/B and PS/B, freezing of the ESW in the standby trains is precluded by the HVAC system operating between 50° F and 105° F. [[Piping running through tunnels and trenches are below grade so that freezing of the ESW is not a concern. Stagnant and exposed portions of the system are heat traced to ensure that the ESW inside these structures is maintained above 32° F.]]

The ESW piping from the pump discharge after passing through the discharge strainers ~~drops underground and~~ runs to the PS/Bs and reactor building through the ~~. The~~ ESW tunnels near the building ~~are located at -26' 4'' elevation~~. After serving the CCW HXs and the essential chiller units ESW piping runs to the UHS.

The COL Applicant is to determine the piping layout of the UHS to maintain the ESWS/UHS pressure above saturation pressure for all operating modes. [[The piping layout of the UHS maintains the ESWS/UHS system pressure downstream of the pump discharge check valve above their saturation pressure at 140° F design temperature by ensuring that no piping high points are above the cooling tower spray header.]] This prevents potential void formation during pump stoppage. During pump operation, due to the addition of the dynamic head to the static head, the ESWS/UHS system pressure will be above saturation pressure.~~The ESWPs are sized to provide positive pressure at the highest point in the system. The system is designed for 140° F.~~ The system layout and the design assure that the fluid pressure remains above saturation conditions at all locations during all modes of operation.

~~The ESWS layout ensures that the fluid pressure in the system is above saturation conditions at all locations.~~ The ESWS layout, in combination with the motor-operated valves (MOVs) at the discharge of each ESWP, minimizes the potential for transient water hammer. The starting logic of the ESWP interlocks the operation of the motor operated valve with the pump operation. ~~The v~~[[Voiding in any train due to potential ESW drain down through the cooling tower spray nozzles may occur ~~on~~ during loss of offsite power and subsequent pump trip.]] To preclude water hammer on pump re-start, the MOV at each pump discharge is interlocked to close when the pump is not running or is tripped. This interlock prevents the pump from starting if the valve is not closed except during emergency situations such as an accident or LOOP events. Upon receiving the pump actuation signal such as an ECCS actuation or LOOP sequence signal~~. After a predetermined time delay after the pump starts~~, the MOV starts to gradually open to preclude water hammer. The ESWP and ESWP discharge MOV interlock is overridden

The COL Applicant is ~~to develop operating procedures~~ to verify system layout ~~and performance~~ of the ESWS and UHS and is to develop operating procedures to assure that the ESWS and UHS ~~is~~are above saturation conditions ~~throughout the system~~for all operating modes.

The COL ~~a~~Applicant is to develop maintenance and test procedures to monitor debris build~~-~~up and flush out debris.

~~The COL Applicant is to provide the piping, valves and other design related to the site specific UHS.~~

9.2.1.2.2 Component Description

Table 9.2.1-1 shows the design parameters of the major components in the system.

9.2.1.2.2.1 ESWPs

Four 50% capacity ESWPs, one per train, supply cooling water to remove heat from the recipient components, and then discharge the heated water to the UHS. Approximately 12,043 gpm ESWP flow is required for all modes of plant operation as indicated in the DCD Table 9.2.1-4. This provides approximately 7.7 percent margin to the design ESWP flow rate of 13,000 gpm. The margin allows for pump and heat transfer degradation by fouling, leakages, excessive pressure drop across system components or, fluctuations due to supplied electrical frequency.

The pumps are powered from the Class 1E ~~normal~~ ac power system. On loss of offsite power, the pumps are automatically powered from their respective emergency power source.

Each pump is designed to provide 13,000 gpm flow at the required total dynamic head. The required pressure drop across the ESWS components and piping (within standard plant design scope) is approximately 100 feet. The COL Applicant is to determine the required ESWP TDH by adding pressure drop across the site specific components and piping and maximum static lift to this pressure drop. The COL Applicant is to provide the site specific data for the ESWPs and assure that the selected ESWP will require less NPSH than the minimum available NPSH under all operating conditions. The COL Applicant is to assure that the sum of the shut-off head of the selected ESW pumps and the static head will not result in exceeding the ESWS design pressure. The UHS level is based on the 30-day emergency cooling at design basis accident heat loads, pump(s) operating at design flow rates with maximum cooling water temperature of 95° F. The potential for vortex formation is evaluated and the available NPSH computed using these parameters. The COL Applicant is to evaluate the potential for vortex formation based on the most limiting assumptions that apply (e.g., temperature, flow rate, operation of other pumps for vortex evaluation).

The ~~ESWP motors are water cooled~~ mode of cooling of the ESWP motors is site-specific and shall be determined by the COL Applicant.

9.2.1.2.2.2 Strainers

- The CCWS, in conjunction with the ESWS, is capable of maintaining the outlet temperature of the CCW heat exchanger below the limits of 110 °F during a design basis accident with loss of offsite power.

9.2.2.1.2 Power Generation Design Bases

The CCWS is designed to:

- Serve as an intermediate system between components containing radioactive fluids, which are cooled by the system, and the ESWS so as to prevent direct leakage of radioactive fluid into the environment through the ESWS.
- Provide sufficient cooling capacity for the components required during normal operating conditions such as normal power operation, normal shutdown and refueling as described below.
- Detect leakage of radioactive material into the system and control leakage of radioactive material out of the system.
- Prevent long term corrosion that may degrade system performance.

9.2.2.1.2.1 Normal Operation

The CCWS is designed to transfer heat from the plant components required to support normal power operation with one train (pump and heat exchanger) unavailable due to on line maintenance and a single active component failure. The CCWS is sized such that the component cooling water supply temperature to plant components is not more than 100°F. Normal operating heat loads are reactor coolant pump, charging pump, letdown heat exchanger, instrument air, spent fuel pool cooling heat exchanger, sample heat exchanger, seal water heat exchanger, blowdown sample cooler, B.A. evaporator, waste gas compressor, and so on. The CCWS provides sufficient surge tank capacity below the low level alarm to allow for operators to take action.

9.2.2.1.2.2 Normal Plant Cooldown

The CCWS is designed to remove both decay and sensible heat from the core and the reactor coolant system in addition to some normal operating heat loads during the latter stages of plant cooldown. The component cooling water system is sized to reduce the temperature of the reactor coolant system from 350°F at approximately 4 hours after reactor shutdown to 140°F using 4 trains while maintaining the component cooling water supply below 110°F. Failure of one train of CCW with another train unavailable due to maintenance will not prevent achieving cold shutdown conditions. The CCWS continues to provide cooling water to the residual heat removal system throughout the shutdown after cooldown is complete.

9.2.2.1.2.3 Refueling

During refueling, cooling water flow is provided to spent fuel pool heat exchangers to cool the spent fuel pool. For a full core off-load cooling water is also supplied to a normal residual heat removal heat exchanger as part of spent fuel pool cooling. The CCWS maintains the spent fuel pit water temperature below 120°F. System operation is with both CCWS divisions available.

9.2.2.2 System Description

The system flow diagram is shown in Figure 9.2.2-1.

The CCWS is the closed loop system that functions as an intermediate system between the various components cooled by CCWS and the ESWS, (Subsection 9.2.1). The CCWS transfers heat and prevents direct leakage of the radioactive fluid from the components to the ESWS.

The CCWS consists of two independent subsystems. One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four trains. Each train has one CCWP and one CCW HX and provides 50% of the cooling capacity required for safety function.

Electrical power to the CCWS is supplied from Class 1E buses that are backed up by Class 1E power supply so that the system is capable to operate during a loss of off site power.

There is the header tie line between trains A and B, and between trains C and D. The header tie line in each subsystem branches~~is branch off~~ into two loops. See Table 9.2.2-1 for the components supplied by each loop.

Each subsystem is served by one CCW surge tank. The CCW surge tank is installed at the highest point of the system to facilitate system air venting to ensure a water solid closed loop and to provide the net positive suction head at the CCWP suction. In addition, the surge tank accommodates the thermal expansion and contraction of the cooling water and potential leakage into or out of the CCWS.

Deminerlized quality water with corrosion inhibitors is circulated in the CCWS. No outside impurities are expected to be infiltrated in the system, therefore, the CCW filter is not necessary~~. The CCW water the filters to protect the plate type CCW heat exchangers are not deemed necessary and not provided~~. The impacts of non-safety related SSC failures in the CCW system will not adversely affect safety-related SSCs to perform their safety related function since the direct impact of a pipe break in the non-safety portion of the system can be accommodated. The CCW system's safety function will be maintained as a result of the nonsafety-related piping failure, and the indirect impact of the pipe break will not impact any SSC safety function.

9.2.2.2.1 Component Descriptions

The CCWS components are described below. Design parameters for major components of CCWS are provided in Table 9.2.2-2.

9.2.2.2.1.1 CCW HX

The CCW HXs transfer heat from the CCWS to the ESWS. The CCW HXs are plate type. The CCW HXs are designated quality group C as defined in Regulatory Guide 1.26 (Ref. 9.2.11-3), seismic category I, and are designed in accordance with the requirements of the ASME Section III, class 3.

9.2.2.2.1.2 CCWP

The CCWP circulates cooling water through the CCW HX and the components cooled by CCWS.

The pumps are horizontal centrifugal pumps and driven by an ac powered induction motor.

The pumps are designated quality group C as defined in Regulatory Guide 1.26, seismic category I, and are designed in accordance with the requirements of the ASME Section III, class 3.

The pumps are designed in consideration of head losses in the cooling water inlet piping based on full power flow conditions, increased pipe roughness, maximum pressure drop through the system heat exchangers, and the actual amount of excess margin etc.

~~Since the difference of installation elevation between the surge tanks and the pumps is large enough, as NPSH available, there is sufficient margin~~The surge tanks are located at a higher elevation than the pumps to ensure sufficient NPSH margin is available.

9.2.2.2.1.3 CCW Surge Tank

The CCW surge tanks are connected to the suction side of the CCWP. The surge tank accommodates the thermal expansion and contraction of the cooling water and potential leakage into or from the CCWS. Makeup water is supplied to the respective surge line.

The CCW surge tank is designated quality group C as defined in Regulatory Guide 1.26, seismic category I, and is designed to the requirements of the ASME Section III, class 3.

In case of a small leak out of the system, makeup water is supplied as necessary until the leak is isolated.

The makeup water can be supplied from the following systems:

- Demineralized water system (DWS) which supplies the demineralized water
- Primary makeup water system (PMWS) which supplies the deaerated water and primary makeup water
- Refueling water storage system (RWS) which supplies the refueling water

Deaerated water is used for initial filling of this system and demineralized water is used for automatic makeup when the tank water level reaches a low level setpoint.

If necessary, primary makeup water and refueling water may be used during an emergency. Refueling water storage pit is water source of seismic category I.

Water chemistry control of CCWS is performed by adding chemicals to the CCW surge tank to prevent long term corrosion that may degrade system performance. The CCW in the surge tank is covered with nitrogen gas to maintain water chemistry.

In order to provide redundancy for a passive failure (a loss of system integrity resulting in abnormal leakage), an internal partition plate is provided in the tank so that two separate surge tank volumes are maintained.

The CCW surge tank capacity of 50% is able to receive the amount of inleak from RCP thermal barrier Hx in consideration of isolation time. Regarding the makeup water source of the RWSP to be seismic category I, this makeup water source provides capacity to accommodate system leakage for seven days. Makeup water supply is performed by an operator by locally operating the manual valves. A vacuum breaker is installed ~~in the surge tank as a countermeasure of the negative pressure in a tank at the time of a sudden fall of tank~~ on the surge tank to prevent damaging the tank in the event of a sudden decrease in water level.

9.2.2.2.1.4 Piping

Carbon steel is used for the piping of the CCWS. Piping joints and connections are welded, except where flanged connections are required.

9.2.2.2.1.5 Valves

・Header tie line isolation valve

The function of this motor operated valve is to separate each subsystem into two independent trains during abnormal and accident conditions. This ensures each safety train is isolated from any potential passive failure in the non-safety portion or another safety train of the CCWS. This valve automatically closes at once upon the following signals:

- Low- low water level signal of a CCW surge tank
- ECCS actuation signal and under voltage signal
- Containment Spray signal

Header isolation meets the single failure criteria by incorporating two header tie line isolation valves. The header isolation valves are designed to close within 30 seconds upon a ~~can be attained even if assumming single failure, since there are two header tie line isolation valves. Since a header tie line isolation valve will be closed in about 10 seconds or less, it is satisfactory to isolate by~~ S+UV signal, P signal, ~~and~~or surge tank water low-low level. Then, in order to resume supply of the cooling water to the RCP thermal barrier heat exchanger and the spent fuel pit heat exchanger, the isolation signal can be bypassed and the isolation valves respond.~~valve close signal currently sent is~~

~~made to bypass and the valve is made to open. CCW pumps are designed such that one CCW pump can supply water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) during normal operation. Therefore, the header isolation valves are maintained to be open.~~ In addition, the header isolation valves are opened in order to supply cooling water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) by one CCW pump during normal operation.

- **Containment Spray/Residual Heat Removal Heat Exchanger (CS/RHRS HX) CCW Outlet Valve**

The CCW which is supplied to the CS/RHR heat exchanger is shutoff by the CCW outlet isolation valve during standby. However, this normal closed motor operated valve automatically opens at once upon ECCS actuation signal plus the respective train CCW pump start signal to establish cooling water flow to the CS/RHR heat exchanger.

- **RCP Thermal Barrier HX CCW Return Line Isolation valve**

Two motor operated valves are located at the CCW outlet of the RCP thermal barrier Hx and close automatically upon a high flow rate signal at the outlet of this line in the event of in-leakage from the RCS through the thermal barrier Hx, and prevents this in-leakage from further contaminating the CCWS.

- **CCW Surge Tank Vent Valve and Relief Valve**

The surge tank vent valve opens upon CCW surge tank high pressure and this valve closes when the radiation monitor level exceeds its set point. The surge tank relief valve provides surge tank overpressure protection.

- **Other Relief Valve**

Other relief valves are provided to relieve the pressure buildup caused by potential thermal expansion when equipment is isolated.

- **Containment Isolation Valve**

Containment isolation valves are installed on CCW lines penetrating containment as described in Subsection 6.2.4.

- **Isolation valve between seismic category I portion and non-seismic category I portion**

~~Two air-operated isolation valves are provided in series on each CCW supply line isolation valves are provided on each CCW supply line (A2 and C2) to the components located in the non-seismic category I buildings (turbine building (T/B) and auxiliary building (A/B). These valves close to protect against CCW seismic category I out-leakage through the non-seismic category I portions by automatic closure upon the demand signals~~The CCW system supplies cooling water to components located in the non-seismic Category I buildings (turbine building and auxiliary building). Each CCW supply line (A2 and C2) has two in-series air operated isolation valves. These valves close automatically to isolate the non-seismic Category I portion of the CCW system upon receipt of a S+UV signal, P signal or surge tank low-low level signal.

~~CCW out-leakage through the non-seismic CCW return lines (A2 and C2) is prevented by check valves series located on the return line for components located in the non-seismic Category I buildings (i.e. the turbine (T/B) and auxiliary building (A/B))~~ In-series check valves are provided on the CCW return lines from the non-seismic Category I portion of the CCW system (See Figure 9.2.2-1, Sheet 9 of 9).

The CCW supply header (A2 and C2) isolation valves close automatically when one of the following occurs (See Figure 9.2.2-1, Sheet 9 of 9).

The isolation valves on auxiliary building supply line

- Low- low water level signal of the component cooling water surge tank
- ECCS actuation signal
- Containment spray signal

b) The isolation valves on turbine building supply line

- Low- low water level signal of the component cooling water surge tank
- ECCS actuation signal and under voltage signal
- Containment spray signal

~~· **RCP Thermal Barrier HX CCW Return Line Isolation valve**~~

~~These Valves function to supply cooling water to the RCPs of header A-1 (or C-1) in the event cooling is lost due to a single failure during on-line maintenance of a CCW pump. The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).~~

· **RCP CCW tie line isolation valve**

This normally closed motor operated valve opens when it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance, and ensures the thermal barrier cooling water.

· **RCP motor CCW supply line isolation valve**

This normally open motor operated valve closes when it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance, and ensures the thermal barrier cooling water.

· **RCP CCW supply line isolation valve**

This normally open motor operated valve closes automatically upon P signal to shutoff the component cooling water flow to the containment vessel.

- **RCP CCW return line isolation valve**

This normally open motor operated valve closes to establish the return line of the thermal barrier cooling water in the case it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance. The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).

9.2.2.2.2 System Operations

Table 9.2.2-4 and 9.2.2-5, respectively, provide heat loads and water flow balance for various operating modes.

9.2.2.2.2.1 Normal Power Operation

During normal operation, at least one train from each subsystem is placed in service. A total of two CCWP and two CCW HXs are in operation. A combination of trains in service is trains A or B and trains C or D.

During this operating condition, an operating CCWP in each subsystem supplies CCW to all loops in the particular subsystem with cooling water temperature not exceeding 100 ^{0}F maximum.

CCWPs which are not in service are placed in standby and automatically start upon a low pressure signal of CCW header pressure.

9.2.2.2.2.2 Normal Plant Shutdown

After approximately four hours of normal plant cool down, when the reactor coolant temperature and pressure are reduced to approximately 350 ^{o}F and 400 psig, the standby CCW HXs and pumps are placed in service resulting in four trains (i.e. four CCWPs and four CCW HXs) in operation. The CCWS isolation valve for each of the CS/RHR HXs is opened to supply cooling water to these HXs.

The failure of one cooling train (i.e. failure in one pump or one HX) increases the time for plant cool down, however, it does not affect the safe operation of the plant. The plant can be safely brought to the cold shutdown condition with a minimum of two trains.

During plant cool down by the residual heat removal system, the CCW supply temperature to the various components is permitted to increase to 110 ^{o}F.

9.2.2.2.2.3 Refueling

During refueling, the required number of CCW HXs and pumps is determined by the heat load. Normally, three trains operate in this mode. The remaining train may be taken out of service for maintenance. An operating CCWP in each subsystem supplies CCW to all loops in service in the particular subsystem with a maximum CCW supply water temperature not exceeding 100 ^{o}F.

9.2.2.2.2.4 Loss of Coolant Accident

All CCWP are automatically actuated by ECCS actuation signal. ~~The signal~~The start signal to the ~~pump~~pumps is ~~setting up delay time~~delayed. (Refer to Figure 8.3.1-2 Logic diagrams (Sheet 18 of 24)) The isolation valves for the CS/RHR HXs are automatically opened by the ECCS actuation signal and the same train CCWP start signal. The header tie line isolation valves are closed by an ECCS actuation signal in coincidence with an undervoltage signal, and the CCWS is separated into four individual trains (A, B, C and D). The header tie line isolation valves can be manually reopened from the MCR to restore RCP seal and SFP HX cooling, if required.

As a minimum, two trains are required to operate during a LOCA.

9.2.2.2.2.5 Loss of Offsite Power (LOOP)

In the case of a LOOP, all CCWPs are automatically loaded onto their respective Class 1E power sources. The CCWS continues to provide cooling of the required components.

As a minimum, two trains are required to operate during a LOOP.

9.2.2.2.2.6 Water Hammer Prevention

The CCWS is designed in consideration of ~~the~~ water hammer prevention and mitigation ~~of its~~ in accordance with the following as discussed in NUREG-0927.

- An elevated surge tank to keep the system filled.
- Vents for venting components and piping at all high points in the system.
- After any system drainage, venting is assured by personnel training and procedures.
- System valves are slow acting.

The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer prevention. The procedures should address the operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.

9.2.2.3 Safety Evaluation

The CCWS is designed to perform its safety function with only two out of four trains operating. As shown in Table 9.2.2-3, the CCWS is completely redundant and a single failure does not compromise the system's safety function even if one train is out of service for maintenance.

9.2.2.5.7 CCWP discharge and suction pressure

The CCW pump discharge and suction pressures are locally indicated and are used for CCW pump performance testing.

9.2.2.5.8 CCW supply temperature

The CCW HX outlet temperature is indicated in the MCR. When the temperature exceeds the setpoint, an alarm is transmitted to the MCR.

9.2.2.5.9 Other instrumentation

As shown in Figure 9.2.2-1, the other flow and temperature indicators are provided where required. These indicators are used for initial flow balancing, and flow and temperature verification during plant operation.

9.2.3 [Reserved]

Not applicable to the US-APWR.

9.2.4 Potable and Sanitary Water Systems

[[The objective of the potable and sanitary water system (PSWS) is to provide clean and potable water for domestic use and human consumption and to collect site sanitary waste for treatment, dilution and discharge during normal operation. The system serves all the areas in the T/B, R/B, A/B, access building, firehouse and future facilities.]]

9.2.4.1 Design Bases

[[There are no safety design bases for the potable and sanitary water system. The power generation design bases are as follows:]]

- [[The potable and sanitary water system ~~layout~~ is designed with no interconnection ~~and/or sharing between~~ to systems that could potentially introduce contaminants including radiological contaminates into the system.~~or between the units to prevent contamination due to potential radioactivity or due to backflow making water unfit for human consumption.~~ This conforms to the requirement of GDC 60 (Ref. 9.2.11-1).]]

- [[The potable water is designed to be treated if necessary to prevent harmful physiological effects. Its bacteriological and chemical quality conforms to the requirements of the Environmental Protection Agency "National Primary Drinking Water Standards," 40 CFR 141 (Ref. 9.2.11-4). All state and local environmental protection standards will also be followed, as these may be more stringent than federal requirements.]] The COL Applicant is to confirm that all State and Local Department of Health and Environmental Protection Standards are applied and followed. The COL Applicant is to confirm the source of potable water to the site and the necessary required treatment.

- [[The distribution of the potable water by the PSWS is in compliance to the "Occupational Safety and Health Standard." 29 CFR 1910, 141 (Ref. 9.2.11-5).]]

- [[The supply capacity of potable water is to provide a quantity of potable water based on 20 gal/person/day for the largest number of persons expected to be at the station during a 24-hour period of power generation or outages. The potable water system is designed to provide a storage capacity of 25,000 gallons.]] The COL Applicant is to confirm the source of potable water to the site and the necessary require treatment. The COL Applicant is to determine the total number of people at the site and identify the usage capacity. Based on these numbers the COL Applicant is to size the potable water tank and associate pumps if used.

- [[Water heaters provide hot water to the main lavatory, shower areas, and other locations where needed. The heater capacity is based on providing an adequate supply of hot water for the anticipated maximum drawdown in the plant. The heater also provides a storage capacity equal to the probable maximum hourly demand for hot water.]]

- [[The system maintains a minimum pressure of 20 psig at the furthermost points in the distribution system.]]

- [[The sanitary drainage system is designed to accommodate 20 gallons/person/day for up to 3500 people during 24-hour period. The above number of people onsite is conservatively based on the largest number of people during plant construction. However, during normal operation, as well as during plant refueling outages, the number of people will be considerably less and the sanitary drainage capacity requirements is reduced. This may result in storage tank modifications based on reduced usage.]]

- [[Sanitary drainage from all remote buildings onsite will be directed to individual sump-lift station, supplied with sewerage grinder transport pump for discharge to a treatment facility.]] The COL Applicant is to confirm that the sanitary waste is sent to the onsite plant treatment area or they will use the city sewage system. The COL Applicant is to determine the total number of sanitary lift stations and is to size the appropriate interfaces.

9.2.4.2 System Description

~~For COL applicants using on-site wells for potable water supply, the potable water system described below applies. For any other source of potable water the COL applicant is responsible for the design and supply of the potable water to the site. The sanitary drainage system design applies for any source of potable water supply.~~

9.2.4.2.1 General Description

[[The potable and sanitary water system flow diagram is shown in Figure 9.2.4-1. Major component data are provided in Table 9.2.4-1.]] The COL Applicant is to confirm Table

[[9.2.4.2.2.4 Hot Water Heaters]]

[[Local potable water hot water heaters are used to ~~produce~~ provide hot water to building -specific areas based on their requirements. Potable ~~Water~~ water ~~from the source~~ is supplied directly ~~potable water tank is supplied~~ to the hot water heater, and which is then routed to the shower and toilet areas and to other plumbing fixtures and equipment requiring domestic hot water service. Local electric water heaters are provided as required to serve restricted or possible contaminated areas such as the MCR. Points of use, inline electric water heating elements are used to generate hot water for the MCR and the T/B areas.]]

[[9.2.4.2.2.5 Valves]]

[[Relief valves are provided on all equipment and in all piping requiring pressure relief.]]

9.2.4.2.3 System Operation

[[Water from the deep wells onsite is pumped and stored in the potable water storage tank. Low water level instrumentation in the potable water storage tank generates a signal to start the well water pumps and supply makeup to the potable water system storage tank. High water levels in the potable water system storage tank produce a signal that stops the well water pumps.]]

[[Prior to supply well water entering the potable water system storage tank, supply water is disinfected. A minimum residual chlorine level of 0.5 ppm is maintained in the supply water prior to entering the potable water system storage tank. The chlorination system is activated and deactivated by a flow signal generated by the fill valve located upstream of the potable water system storage tank.]]

[[Two potable water pumps and a jockey pump are used to supply potable water throughout the plant. The potable water system pumps are activated sequentially to maintain adequate pressure throughout the distribution system. A pressure transmitter is provided downstream of the potable water system pumps to control their start/stop sequences. The jockey pump operates continuously to maintain system pressure during low-flow requirement periods.]] The COL Applicant is to identify the potable water supply and describe the system operation.

Potable water is supplied to areas that have the potential for contamination with radioactivity. Where this potential for contamination exists, the potable water system is protected by installing backflow prevention device.

[[No interconnections exist between the potable water system and any system using water for purposes other than domestic water service, including any potentially radioactive system. The water supply from the other sources supplying water to potentially radioactive systems is designed to use an air gap to prevent contamination of the potable water system from other systems.]]

9.2.4.3 Safety Evaluation

The ultimate heat sink (UHS) consists of an assured source of water with associated safety-related structures designed to dissipate the heat rejected from the ESWS during normal and accident conditions. UHS peak heat loads and long term heat loads are shown in Tables 9.2.5-1 and 9.2.5-2, respectively. The UHS system is safety-related and designed to meet the requirements of Regulatory Guide 1.27 (Ref. 9.2.11-2).

9.2.5.1 Design Bases

~~The UHS is designed to dissipate the maximum possible total heat load, including that of a LOCA under the worst combination of adverse environmental conditions, even freezing, and cool the unit for a minimum of 30 days (or 36 days for cooling pond in accordance with Regulatory Guide 1.27) without makeup water. The UHS is designed for a single nuclear power unit and is not shared between units.~~ The UHS is a site-specific interface with the ESWS. The design information provided in this subsection establishes interface requirements applicable to the UHS design and to be provided by the COL Applicant based on specific site characteristics including meteorological data.

The UHS designed in accordance with GDC 44 ~~bases~~ to ~~meet the safety-related functional requirements are provided below~~:

- Dissipate the maximum total heat load from the ESWS under normal and accident condition, including that of a LOCA or safe shutdown scenario with LOOP under the worst combination of adverse environmental conditions, even freezing, and cool the unit for a minimum of 30 days (or minimum of 36 days for cooling pond) in accordance with Regulatory Guide 1.27 without makeup water. The decay heat is estimated using ANSI/ANS 5.1, “Decay Heat Power for Light Water Reactors” (Ref. 9.2.10-6).

- Provide suitable component redundancy such that the system’s safety functions can be performed in the event of a single active component failure, coincident with an accident such as a LOCA and safe shutdown with LOOP under extreme meteorological conditions, using either offsite power or onsite emergency power sources.

- Provide the capability to isolate components, systems or piping such that safety functions are not compromised.

The UHS is designed for a single nuclear power unit and is not shared between units (GDC 5). The UHS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including LOCA. These environmental effects include dynamic effects that may result from equipment failures or external events (GDC 4).

The UHS is designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform

- 3.3, Wind and Tornado Loads
- 3.4, Water Level (Flood) Design
- 3.5, Missile Protection
- 3.7, Seismic Design
- 3.8, Design of Category I Structures

Site-specific UHS design features to address limiting hydrology-related events are addressed as required by DCD Section 2.4, Hydrologic Engineering (specifically Subsections 2.4.8, 2.4.11 and 2.4.14). The UHS is sufficiently sized to accept the heat rejected (Table 9.2.5-1) from the ESWS.

The heat loads for LOCA and safe shutdown conditions with LOOP for up to 36 days are provided in Table 9.2.5-2. These heat loads establish the basis for minimum allowable UHS cooling water inventory and maximum temperature limits, to maintain the ESW supply water temperature less than or equal to 95 ºF under design basis heat load conditions. [At the minimum water level following UHS operation in the limiting scenario in Table 9.2.5-2, ESW pump NPSH requirements are met without assuming UHS inventory make-up from external sources, in accordance with RG 1.27.] Technical Specifications by the COL Applicant (COL 16.1_3.7.9 (1)) prescribe operating limits for [minimum UHS usable water capacity and maximum UHS initial water temperature.

[Table 9.2.5-4 on Failure Modes and Effects Analysis (FMEA) of the UHS concludes that no single failure, coincident with one train being unavailable due to maintenance and a loss of offsite power compromises the safety functions.

9.2.5.4 Inspection and Testing Requirements

The COL Applicant will provide test and inspection details based on type of UHS to be provided. These details will include inspection and testing requirements necessary to demonstrate that fouling and degradation mechanisms are adequately managed to maintain acceptable UHS performance and integrity.

The UHS supports ESWS operation and is therefore tested as part of the ESWS preoperational test described in Subsection 14.2.12.1.34. As indicated in Subsection 14.2.12, the COL Applicant is responsible for testing outside the scope of the certified design.

Periodic inspections and tests of UHS and ESWS components and subsystems are performed to verify proper operation and system operability. This includes ASME Section XI requirements as discussed in Section 6.6.

The COL Applicant is to develop maintenance and test procedures to monitor debris build up and flush out debris.

9.2.9.1.2 Power Generation Design Bases

The following is a list of the non-safety power generation design bases:

- The non-ESW system provides cooling water to the TCS HXs located in the T/B.
- The non-ESW system is designed to transfer heat to the CWS during all modes of plant operation.
- The non-ESW system is designed with redundant components available during all modes of normal power operation.

9.2.9.2 System Description

9.2.9.2.1 General Description

The non-ESW system is a once through system that draws water from the circulating water piping at the condenser inlet and returns water to the CWS piping at the condenser outlet after passing through the TCS HXs.

The system is composed of three 50% capacity non-ESW system pumps, three 50% capacity TCS HXs, two 100% capacity strainers, piping, valves, controls and instrumentation.

Figure 9.2.9-1 shows the non-ESW system configuration. Equipment and component classification and applicable codes and standards are provided in section 3.2

The non-ESW pumps located in the T/B, take suction from the circulating water piping and pumps water through the strainers and the TCS HXs to a common discharge header. The heat from TCS is removed in the HXs by non-ESW system and the heated water is returned to the cooling tower through the circulating water piping.

The non-ESW system is arranged such that any two of three pumps can operate in conjunction with any two of three TCS HXs to meet system flow requirements. One out of two 100% capacity strainers is used. The pumps take suction from a common header and the discharge flows from the HXs combine into a common discharge header.

Each heat exchanger is provided with two separate isolation valves and piping around the heat exchanger for back flushing. One valve is located in the piping from the inlet of the heat exchanger inlet isolation valve to the inlet of the heat exchanger discharge isolation valve and the second valve is located in the piping from the outlet of the heat exchanger inlet isolation valves to the outlet of the heat exchanger discharge isolation valve. To initiate manual back flush operation, both bypass valves are opened and the heat exchanger isolation valves~~vavles~~ are closed. Cooling water flows from the discharge side into the heat exchanger and discharged ~~form~~from the heat exchanger inlet side to the service water discharge header.

The temperatures in the system are moderate and the fluid pressure in the system is kept higher than the above saturation conditions at all locations in the system. This along

~~the safety evaluation.~~ The COL Applicant is also to design the pipes entering and exiting the pipe tunnel based on the location of the UHSRS.

COL 9.2(8) The COL Applicant is to specify the following ESW chemistry requirements:

- *A chemical injection system to provide non-corrosive, non-scale forming conditions to limit biological film formation.*
- *Type of biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor and silt dispersant based on the site conditions.*

COL 9.2(9) COL Applicant is to confirm the storage capacity and usage of the potable water.

COL 9.2(10) COL Applicant is to confirm that all State and Local Department of Health and Environmental Protection Standards are applied and followed.

COL 9.2(11) COL Applicant is to identify the potable water supply and describe the system operation.~~The COL Applicant is to confirm the source of potable water to the site and the necessary required treatment.~~

COL 9.2(12) COL Applicant is to confirm that the sanitary waste is sent to the onsite plant treatment area or they will use the city sewage system.

COL 9.2(13) ~~COL Applicant is to identify the potable water supply and describe the system operation.~~Deleted

COL 9.2(14) COL Applicant is to confirm Table 9.2.4-1 for required components and their values.

COL 9.2(15) The COL Applicant is to determine the total number of people at the site and identify the usage capacity. Based on these numbers the COL Applicant is to size the potable water tank and associated pumps.

COL 9.2(16) ~~The COL Applicant is to provide values to the component Table 9.2.4-1 based on system and component descriptions from Section 9.2.4.2.1 and 9.2.4.2.2 respectively.~~Deleted

COL 9.2(17) The COL Applicant is to determine the total number of sanitary lift stations and is to size the appropriate interfaces.

COL 9.2(18) The COL Applicant is to determine the type of the UHS based on specific site conditions and meteorological data.

COL 9.2(19) The COL Applicant is to design the UHS to receive its electrical power supply, if required by the UHS design, from safety busses so that the safety functions are maintained during LOOP. The UHS also receives its standby electrical power from the onsite emergency power supplies during a LOOP.

COL 9.2(22) The COL Applicant is to provide results of UHS capability and safety evaluation of the UHS based on specific site conditions and meteorological data. The COL Applicant is to use ~~at least 30 years~~ site specific meteorological data and heat loads data for UHS performance analysis per Regulatory Guide 1.27.

COL 9.2(23) The COL Applicant is to provide test and inspection requirements of the UHS. These ~~is to~~ include inspection and testing requirements necessary to demonstrate that fouling and degradation mechanisms are adequately managed to maintain acceptable UHS performance and integrity.

COL 9.2(24) The COL Applicant is to provide the required alarms, instrumentation and controls details based on the type of UHS to be provided.

COL9.2(25) The COL ~~a~~Applicant is to ~~will~~ develop ~~operating and maintenance procedures for the~~system filing, venting, keeping the system full, and operational procedures to minimize the potential for water hammer; to analyze the system for water hammer impacts; to design the piping system to withstand potential water hammer forces; and to analyze inadvertent water hammer events ~~ESWS to address water hammer issues~~ in accordance with NUREG-0927.

COL 9.2(26) The COL ~~a~~Applicant is to specify appropriate sizes of piping and pipe fittings such as restriction orifices to prevent potential plugging due to debris buildup,and develop maintenance and test procedures to monitor debris build up and flush out debris.

COL 9.2(27) The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer prevention.

COL 9.2(28) The COL Applicant is to provide the piping, valves, materials specifications, and other design details related to the site specific UHS.

COL 9.2(28) The COL Applicant is to provide the piping, valves, materials specifications, and other design related to the site specific UHS.

COL 9.2(29) The COL Applicant is to provide the safety evaluation of the capability of the ESWS to: (1) isolate its site-specific, nonsafety-related portions; and (2) provide measures to prevent long-term corrosion and organic fouling that may degrade its performance, per Generic Letter (GL) 89-13.

COL9.2(30) The COL Applicant shall conduct periodic inspection, monitoring, maintenance, performance and functional testing of the ESWS and UHS piping and components, including the heat transfer capability of the CCW heat exchangers and essential chiller units, consistent with GL 89-13 and GL 89-13 Supplement 1. The COL Applicant is to develop operating procedures to periodically alternate the operation of the trains to ensure performance of all trains is regularly monitored.

COL9.2(31) The COL Applicant is to verify the system layout of the ESWS and UHS and is to develop operating procedures to assure that the ESWS and UHS are above saturation conditions for all operating modes.

COL 9.2(32) The COL Applicant is to provide a void detection system with alarms to detect system voiding.

COL9.2(33) The COL Applicant is to provide the design details of the strainer backwash line, vent line, and their discharge locations.

9.2.11 References

9.2.11-1 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

9.2.11-2 Ultimate Heat Sink for Nuclear Power Plants, Regulatory Guide 1.27 Revision 2, January 1976.

9.2.11-3 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, NRC Regulatory Guide 1.26 Revision 4, March 2007.

9.2.11-4 National Primary Drinking Water Standards, Environmental Protection Agency, Title 40, Code of Federal Regulations, 40CFRPart 141.

9.2.11-5 Occupational Safety and Health Standard, Occupational Safety and Health Administration, Department of Labor, Title 29, Code of Federal Regulations, 29CFRPart 1910.

9.2.11-6 American Nuclear Society Standards Committee Working Group, American National Standard for Decay Heat Power in Light Water Reactors, ANS 5.1, August 1979.

9.2.11-7 Electric Power Research Institute Palo Alto, California, Advanced Light Water Reactor Utility Requirements Document, Rev.8.

9.2.11-8 ANSI B31.1 Power Piping Code.

Table 9.2.2-5 Component Cooling Water system Flow Balance Unit of Flow Rate [gpm]

Train	Normal Power Operation	Cooldown by CS/RHRS	Accident	Safe Shutdown
A & B	600	9400	4700	4700
A1	4575	4575	4575	4575
A2	~~1948~~1988	~~1948~~1988	310	310
Subtotal	~~7123~~7163	~~15923~~15963	9585	9585
C & D	600	9400	4700	4700
C1	4575	4575	4592	4575
C2	925	1490	0.0	0.0
Subtotal	6100	15465	9292	9275
The total number of operating CCW pumps	2	4	2	2

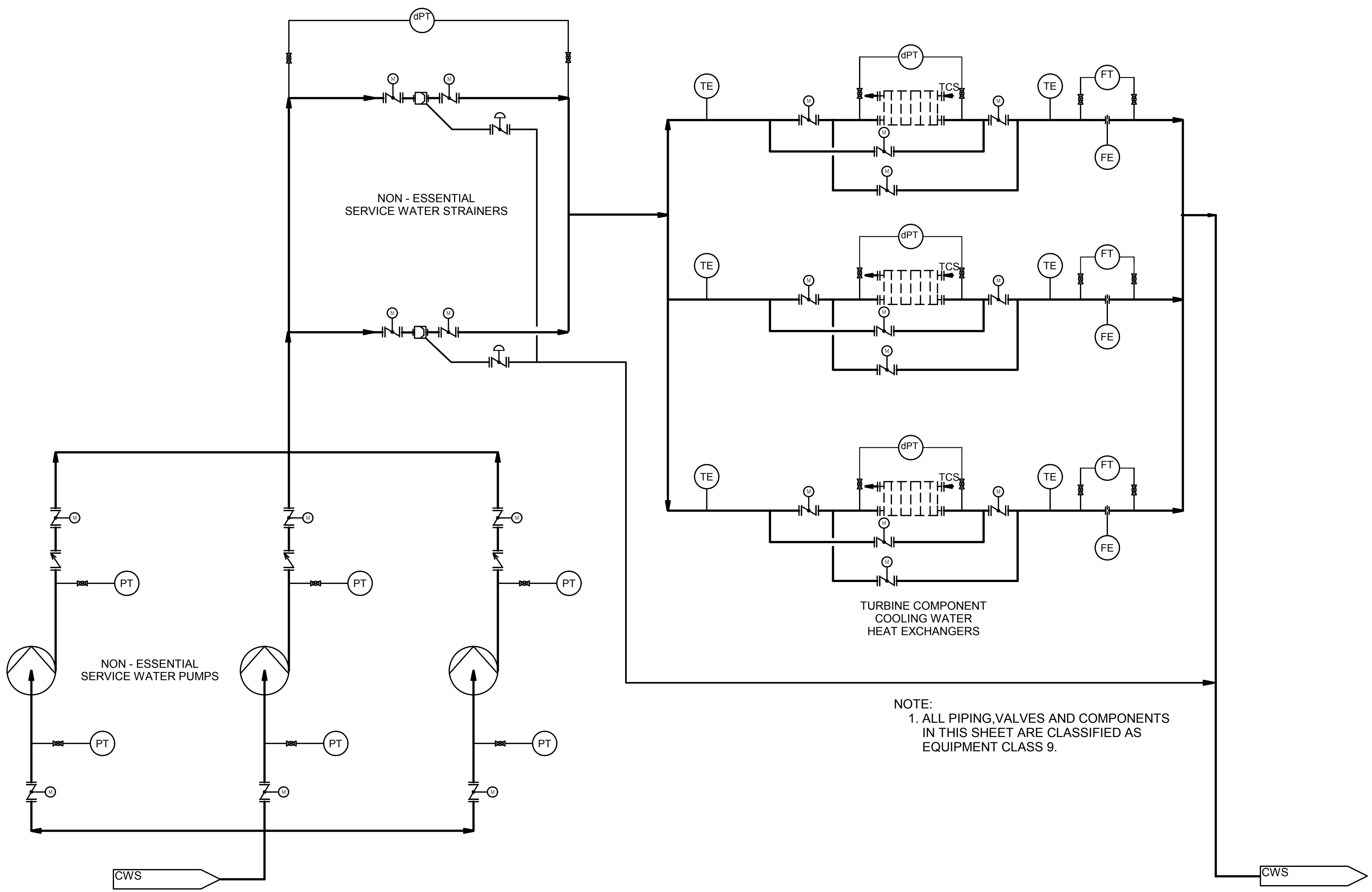


Figure 9.2.9-1 Non-Essential Service Water System Flow Diagram

determine pH and conductivity levels, dissolved oxygen, residual oxygen scavenger, silica, chloride, sodium, and sulfate. These measurements are used to control water chemistry and to permit appropriate corrective action by the plant operating staff. In addition, grab sample capabilities are provided at each of these monitoring points to analyze other chemicals.

The purpose of the SSS is to provide the data necessary for controlling the water quality of the secondary plant systems listed in Table 9.3.2-4. The SSS is located in the T/B.

The SSS samples are specified in Table 9.3.2-4. Primary coolers are provided for the samples whose temperatures exceed 125 ^{o}F. All samples are conditioned to 115 ^{o}F by cooling water or constant temperature bath to approximately 40 psig by pressure regulators.

Sample points may be used to continuously monitor representative samples. The sample line and sample sink drain into the SSS are collected and drained to the T/B floor sump. Each sample line has a grab sampling capability for laboratory analysis. Sampling point is also provided circulating water system (CWS) to ensure that no harmful effects will result to CWS piping and valves due to improper water chemistry.

9.3.2.2.5 Steam Generator Blowdown Sampling System

The SGBDSS is provided to control the steam generator (SG) secondary side water quality and to detect a leak or failure of SG tubes. The SGBDSS includes blowdown lines and blowdown sample lines. The SGBDSS also includes blowdown sample coolers, pressure reducing valves, a radioactive process monitor, instruments, piping and valves. The SGBDSS components are described in Chapter 10, Subsection 10.4.8. The sample points are discussed in Table 9.3.2-5.

The SGBDSS performs the following functions:

- Monitors secondary water quality in SGs to maintain acceptable secondary coolant water chemistry.
- Detects primary to secondary SG tube leakage.
- Containment isolation.

Based on radioactivity and water chemistry monitoring, the blowdown water is purified by its own polishing system. Otherwise, the blowdown is sent to the [[waste water system (WWS)]] when disposal is required due to its chemical content; or during start-up. Blowdown water can also be sent to the LWMS when disposal is required due to radioactivity content.

Blowdown sample water is passed through the blowdown sample coolers and pressure reducing valves for sampling. Blowdown samples are used periodically to check the Choride and Sulfate of the SG secondary water, continuously to monitor the specific conductivity, cation conductivity, pH, sodium of the SG secondary water in order to maintain acceptable secondary coolant water chemistry and continuously for monitoring radiation in order to detect leakage or failure of a SG tube.~~Blowdown samples are used periodically to check the conductivity and pH of the SG secondary water and continuously monitored for radiation to detect leakage or failure of a SG tube.~~

All SG blowdown lines and SG blowdown sample lines are automatically isolated from the containment on any signal of automatic initiation of the emergency feedwater pumps and/or a high radiation signal of the radiation monitors.

9.3.2.2.6 Manual Local Grab Sample Provisions

Local grab sampling points, as listed in Table 9.3.2-6, are provided as needed for various processes. Manual grab sample points are provided for the liquid sample points as required by the operator. Quick-disconnect type couplings are used for sample vessel connections to provide a convenient and expeditious way of sampling. Liquid tanks are stirred using pumps in recirculation mode, and stir nozzle mixing devices in order to enable collection of a representative sample. The tank sample point is located at the discharging line of the pump to allow the operator to take a well mixed sample of the stirred liquid. The inner diameter of process and sampling piping is selected to maintain turbulent flow under normal operating flow rate and accordingly this feature prevents suspended solids from sedimentation and plate out.

Grab sample points for liquids are identified in Table 9.3.2-6. Grab sample points are indicated on the appropriate system flow diagrams.

9.3.2.3 Safety Evaluation

Except for the associated containment penetrations, the process and post-accident sampling systems do not have a safety function. Chapter 6, Subsections 6.2.4 provides the safety evaluation for the containment isolation system. All PLSS, PGSS, PASS and SGBDSS lines penetrating the containment can be isolated at the containment boundary by valves that close either upon receipt of a containment isolation signal or by manual actuation. (Chapter 6, Subsection 6.2.4 provides a detailed discussion of containment isolation)

9.3.2.4 Inspection and Testing Requirements

Proper operation of the process and post-accident sampling system is initially demonstrated during preoperational testing.

The proper operation and availability of the PLSS, PGSS, SSS and SGBDSS are proven in service by their use during normal plant operation. Samples from the PLSS, PGSS and PASS are drawn manually for laboratory analysis. The results of this analysis are checked by calibrating the laboratory instruments against known compositions or check sources.

The SSS and SGBDSS draw continuous samples from the turbine component cooling water system for monitoring water quality. The operation of the SSS and SGBDSS is verified by observing that continuous sample flow is maintained through the analyzers. The calibration of the analyzers is checked periodically by auto-calibration features on the

9.3.4.1.1 Safety Design Bases

The safety design bases for the CVCS are as follows:

- Provide reactor coolant pressure boundary (RCPB).
- Provide containment isolation of CVCS lines penetrating containment.
- Provide capability for isolation the charging line upon ECCS actuation ~~safety injection~~ signal and high Pressurizer water level.
- Provide isolation capability for a boron dilution source in reactor coolant to prevent inadvertent RCS boron dilution.

9.3.4.1.2 Power Generation Design Bases

The power generation design bases for the CVCS are as follows:

- Maintain appropriate volume and quality of reactor coolant for the RCS
- Regulate the boron concentration for the chemical shim control
- Remove fission products and ionic corrosion products from the reactor coolant
- Supply seal water to the reactor coolant pump seals
- Receive borated water discharged from the RCS
- Provide pressurizer auxiliary spray water for depressurization of the RCS when none of the RCPs are operating

System reliability is achieved by the use of redundant equipment (pumps, filters, and demineralizers). The equipment classification for the CVCS is contained in Chapter 3, Section 3.2.

9.3.4.1.2.1 Reactor Coolant System Inventory Control and Makeup

The CVCS provides a means to maintain a programmed inventory of reactor coolant during all phases of plant operation.

The CVCS is capable of maintaining a constant volume in the RCS by means of a continuous feed and bleed process. The nominal makeup and letdown flowrates are shown in Table 9.3.4-2. The amount of feed is automatically controlled based on pressurizer water level. The amount of bleed is selected by switching the proper combination of the letdown orifices in the letdown flow path to accommodate the various plant operating conditions. The CVCS has sufficient makeup capacity to maintain the minimum required inventory in the event of minor leaks in the RCS, as discussed in Subsection 9.3.4.2.7.4.

maintained until the charging pumps are powered from an alternate power source and seal water injection restarts.

9.3.4.2 System Description

The CVCS consists of charging pumps, regenerative heat exchanger, letdown heat exchanger, excess letdown heat exchanger, demineralizers, filters, pumps, tanks, and associated valves, piping, and instrumentation. The system parameters are given in Table 9.3.4-2. The piping and instrumentation diagram for the CVCS is included in Figure 9.3.4-1. The seismic category and quality group classification for CVCS components are specified in Chapter 3, Section 3.2.

9.3.4.2.1 Reactor Coolant System Inventory Control, Reactor Coolant Pump Seal Injection and Makeup

Reactor coolant is discharged to the CVCS from the crossover piping. During normal operation, the reactor coolant is cooled by flowing through the shell side of the regenerative heat exchanger, and then flows through the letdown orifices where the reactor coolant pressure is reduced. The coolant passes through the letdown heat exchanger, where its temperature is further reduced. The reactor coolant pressure is further reduced by a pressure control valve located downstream of the letdown heat exchanger. This valve is provided to maintain upstream pressure to prevent flashing downstream of the letdown orifices.

Normally, the reactor coolant flows through one mixed bed demineralizer inlet filter and one mixed bed demineralizer, then passes through the reactor coolant filter, and then enters the volume control tank (VCT) through the spray nozzle.

The gas space of the VCT is filled with hydrogen. The hydrogen pressure in the VCT is controlled to establish the concentration of hydrogen dissolved in the reactor coolant.

To reduce, if required, the amount of the radioactive gases dissolved in the reactor coolant, if required, the gas in the VCT gas space can be purged to and processed by the gaseous waste management system (GWMS).

Normal charging is performed by utilizing a single charging pump. The charging pump takes suction from the VCT and returns the purified reactor coolant to the RCS. The flow rate of the charging pump is controlled by the flow control valve located in the charging line and the flow control valve located in the reactor coolant pump seal injection line. The charging line flow control valve is controlled by the charging flow rate control unit, which is adjusted by the pressurizer water level signal, the charging flow rate signal and the letdown flow rate signal. A portion of the flow is directed to the reactor coolant pumps through a seal water injection filter. The flow to the reactor coolant pumps is controlled by a flow control valve located in the reactor coolant pump seal injection line. The flow control valve in the seal injection line is adjusted by a reactor coolant pump seal injection flow rate signal.

A minimum amount of flow branches off at the discharge side of the charging pump for pump protection, and returns to the outlet of the VCT through the seal water heat exchanger.

Most of the charging flow is injected to a cold leg of the RCS through the tube side of the regenerative heat exchanger. The regenerative heat exchanger performs heat exchange between the charging flow and the letdown flow to raise the charging flow temperature approximately to the temperature in the reactor coolant loop.

A branch line off the charging line downstream of the regenerative heat exchanger is routed to the auxiliary pressurizer spray line. The auxiliary pressurizer spray provides a mean of cooling and depressurizing the pressurizer near the end of plant cooldown, when the reactor coolant pumps are not operating.

The remainder of the charging flow is supplied to the RCP shaft No. 1 seal through the seal water injection filter. A portion of the seal water flows along the pump shaft downward into the RCS through the pump shaft bearing-labyrinth seal and the thermal barrier. The remainder of the seal water runs along the pump shaft upward through the No.1 seal and exits the pump and discharges to the common No. 1 seal water return line. The seal water exits the containment vessel, passes through the seal water return strainer and the seal water heat exchanger, and returns to the VCT outlet line.

The excess letdown line from the RCS is provided for the possible malfunction of the normal letdown line. The reactor coolant is directed to the CVCS from the crossover piping to the tube side of the excess letdown heat exchanger where it is cooled to about 165 °F. The excess letdown flow rate is controlled by the excess letdown flow control valve located downstream of the heat exchanger. During excess letdown operation, the flow joins with that from the No. 1 seal water flow return line, and flows through the seal water heat exchanger, then on to the outlet line of the VCT. The excess letdown flow can also be discharged directly to the reactor coolant drain tank.

The excess letdown flow path is also utilized to supplement the normal letdown flow at the final stage of the plant heatup.

Excess reactor coolant due to reactor coolant expansion during heatup of the RCS can be released through the excess letdown flow path that drains into the reactor coolant drain tank.

Surges due to load changes in the RCS are mostly accommodated by the pressurizer; however, the VCT provides surge capacity for part of the reactor coolant expansion volume which can not be accommodated by the pressurizer. The letdown flow normally flows into the VCT. When the water level in the VCT reaches the high-level setpoint, the letdown flow is routed to the holdup tank by the VCT inlet three-way valve. When the water level in the VCT reaches the low-level setpoint, the reactor makeup water control system starts to provide makeup. If the reactor makeup water control system cannot supply sufficient makeup water necessary to prevent decrease of the VCT water level, a low-low VCT level alarm is actuated and the suction of the charging pump is switched from the VCT to the RWSAT. The charging pump can also take suction from the SFP as a SSE makeup water source.

One regenerative heat exchanger is provided. This heat exchanger is used to recover heat from the letdown flow during normal operation and heat up the charging flow to provide increased thermal efficiency and reduce thermal stresses on the pipe nozzle connecting to the RCS. The heat exchanger reduces the letdown flow temperature to prevent steam flashing downstream of the letdown orifices.

The charging flow passes through the tube side, and the letdown flow passes through the shell side. This arrangement allows the shell side to have a lower design pressure than that of the tube side.

The regenerative heat exchanger employs stainless steel materials and all-welded structure.

9.3.4.2.6.5 Letdown Heat Exchanger

One horizontal U-tube letdown heat exchanger is provided. The heat exchanger is designed to cool the letdown flow from the regenerative heat exchanger outlet temperature to the desired operating temperature of the VCT. The letdown heat exchanger outlet temperature is controlled by adjusting the component cooling water flow rate with the temperature control valve placed on the component cooling water outlet line (See Section 9.2.2, component cooling water system).

The letdown flow enters the letdown heat exchanger through the stainless steel tubes, and component cooling water flows through the shell, which is made of carbon steel.

9.3.4.2.6.6 Excess Letdown Heat Exchanger

The excess letdown heat exchanger is designed to cool excess letdown flow equivalent to that portion of the nominal seal injection flow which flows into the RCS through the reactor coolant pump shaft bearing labyrinth seals. The excess letdown is used when the normal letdown line is not available. The heat exchanger is also utilized to supplement maximum letdown flow in conjunction with the normal letdown during the final stages of heat-up.

The letdown water flows through the tube side and component cooling water flows through the shell side of the heat exchanger .

All parts of the heat exchanger in contact with the reactor coolant are made of stainless steel. The shell side is made of carbon steel and has all-welded structure.

9.3.4.2.6.7 Seal Water Heat Exchanger

The seal water heat exchanger is provided to cool reactor coolant from the following sources and discharges to it the charging pump suction:

- Seal water return flow from the reactor coolant pumps
- Letdown flow from the excess letdown heat exchanger

9.3.4.2.7.3 Plant Shutdown

During plant shutdown, when the RHR system is in operation, the RHR system provides reactor coolant to the CVCS, upstream of the letdown heat exchanger in the letdown line. Cooling of the pressurizer fluids can be accomplished by charging through the auxiliary spray connection as an alternative way while pressurizer spray is normally used.

When the purification flow is increased, two charging pumps can be in operation. The letdown flow passes through the letdown heat exchanger, two mixed bed demineralizer inlet filters, two mixed bed demineralizers, two reactor coolant filters, two spray nozzles, and into the volume control tank. During plant shutdown, the gas space of the volume control tank is replaced with nitrogen. The reactor coolant is returned to the RCS through the normal charging flow path.

9.3.4.2.7.4 Reactor Coolant System Leak

One CVCS charging pump is capable of maintaining normal RCS inventory with small system leak if the leakage rate is less than that form a break of a pipe 3/8 inch ~~in~~ inside diameter.

9.3.4.2.7.5 Abnormal Operations

In the case of malfunction of the normal letdown line, reactor coolant is directed from the crossover piping to the tube side of the excess letdown heat exchanger, where it is cooled to about 165 °F. The excess letdown flow rate is controlled by the excess letdown flow control valve located downstream of the heat exchanger.

9.3.4.2.7.6 Boron Dilution Events

The CVCS is designed to provide isolation to limit boron dilution by closing either one of the two redundant safety-related, motor operated valves from the primary makeup water system.

During “dilute” mode and “alternate dilute” mode, a pre-selected quantity of reactor makeup water is supplied to the RCS at a pre-selected flow rate.

When the preset quantity of reactor makeup water has been supplied, the batch integrator will cause the primary makeup water pump to stop and the reactor water control valve to close. The “dilute” and “alternate dilute” modes of operation may be manually terminated at any time by selecting the makeup stop.

When the reactor makeup water flow exceeds the predetermined setpoint, the instruments provide a high alarm signal, the automatic isolation valves close, and the operation is terminated to prevent abnormal boron dilution.

9.3.4.3 Safety Evaluation

The CVCS has redundant, safety-related isolation valves to support the RCPB, charging isolation, and inadvertent boron dilution prevention. The CVCS lines that penetrate containment incorporate valves and piping arrangements that meet the containment isolation criteria described in subsection 6.2.4. Containment isolation valves in the CVCS are required to operate under accident conditions to provide containment isolation,as required.

Since the CVCS supplies non-borated water to the RCS, the potential for inadvertent boron dilution events exists. The design feature for preventing an inadvertent boron dilution is described in Subsection 9.3.4.2.7.6.

The charging line is isolated on a ECCS actuation ~~safety injection~~ signal and a Pressurizer high water level signal, to terminate unnecessary RCS makeup that can cause an overfilling of the pressurizer and steam generator overfilling during a steam generator tube rupture.

During a SBO, the reactor coolant pumps seal integrity is maintained until the charging pumps are powered from an alternate power source and seal water injection restarts using the normal seal injection flow path.

The CVCS is designed to provide makeup for minor leaks in the RCS. The makeup capability is limited to the leakage equivalent to a pipe break with 3/8 inch inside diameter.

The CVCS does not provide an ECCS function. ~~Therefore, the provision for a leakage detection and control program in accordance with 10 CFR 50.34 (f) (xxvi) does not apply.~~

CVCS components and piping are compatible with the radioactive fluids they contain and the functions they perform. The equipment classification for the CVCS is contained in Section 3.2.

The CVCS is designed to ensure that the boric acid solution remains soluble. Heat tracing or a heated area with temperature alarms are provided for portions of the system which normally contain 4 wt. % of boric acid solution, to assure that boric acid solution temperature does not go below 65 °F.

The VCT is designed to withstand vacuum conditions to prevent wall inward buckling and failure. The boric acid tanks are provided with vacuum breakers to prevent a vacuum condition. The holdup tanks are provided with sufficient nitrogen gas supply to prevent vacuum condition.

The CVCS is designed in accordance with the requirements of 10 CFR 50, Appendix A, GDCs are GDC 1, 2, 14, 33, 60, and 61.

The protection of safety-related portions of CVCS against natural phenomena and internal missiles is addressed in the following sections in Chapter 3:

Section 3.3, Wind and tornado loadings;

Section 3.4, Water level (Flood) protection;

9.3.4.5.3.3 Demineralizer and Filter Differential Pressure

Differential pressure gauges are provided for the following filters and demineralizers to provide local indication and high alarm:

- Reactor coolant filters
- Mixed bed demineralizers inlet filters
- Seal water injection filters
- Boric acid filter
- Boric acid evaporator feed demineralizer filter
- Mixed bed demineralizers
- Cation bed demineralizer
- Deborating demineralizer
- Boric acid evaporator feed demineralizer

9.3.4.5.3.4 Pumps Discharge Pressure

Instrumentation is located at the following pump discharge lines to provide local indication of the discharge pressure:

- Charging pump
- Boric acid transfer pump
- Boric acid evaporator feed pump

9.3.4.5.3.5 Charging Header pressure

Instrumentation is provided to indicate the charging header pressure and to provide indication in the MCR.

9.3.4.5.3.6 Excess Letdown Heat Exchanger Outlet Pressure

Instrumentation is provided to indicate the pressure of the reactor coolant coming from the excess letdown heat exchanger and provide indication in the MCR.

9.3.4.5.3.7 Volume Control Tank Hydrogen and Nitrogen Supply Pressure

Instrumentation is provides locally to indicate~~f~~ the volume control tank pressure in order to constantly control the hydrogen and nitrogen pressures to be supplied to the volume control tank.

Table 9.3.1-2 Nominal Component Design Data - Instrument Air System

Air Compressors	
Quantity	2
Type	Rotary
Capacity (each)	600 scfm
Design pressure	150 psig
Air Receivers	
Quantity	2
Type	Vertical cylinder type
Capacity, each	230 ft^3
Design pressure	150 psig
Design code	ASME Section VIII (Ref. 9.3.7-9)
Air Dryers	
Quantity	2
Type	Twin-tower desiccant type
Capacity, each	600 scfm
Design pressure	150 psig
Design code	ASME Section VIII (Ref. 9.3.7-9)
Outlet dew point	Below -~~40~~58 °F at 128 psig

Table 9.3.2-5 Steam Generator Blowdown Sampling System Sample Points

Sample Point No.	Sample Point Name	Analysis[b]
1	A-SG Blowdown	Na[c], SC[c~~a~~], CC[c], pH[c~~a~~] Cl[a], SO_4[a~~d~~]
2	B-SG Blowdown	Na[c], SC[c], CC[c], pH[c] Cl[a], SO_4[a] ~~Na, SC[a], CC, pH[a] Cl, SO_4[d]~~
3	C-SG Blowdown	Na[c], SC[c], CC[c], pH[c] Cl[a], SO_4[a] ~~Na, SC[a], CC, pH[a] Cl, SO_4[d]~~
4	D-SG Blowdown	Na[c], SC[c], CC[c], pH[c] Cl[a], SO_4[a] ~~Na, SC[a], CC, pH[a] Cl, SO_4[d]~~

NOTE:
(a) These points are provided with grab sampling capability but are not always continuously monitored.
(b) Symbols used:
SC - specific conductivity
CC - cation conductivity
~~DO - dissolved oxygen~~
(c) Continuous monitoring ~~during startup only.~~
~~(d) Continuous monitoring during normal operation~~

Table 9.3.4-3 Chemical and Volume Control System Equipment Design Parameters (Sheet 1 of 6)

Charging Pumps		
Number of units	2	
Design flow rate	275 gpm	
Type	Multistage horizontal centrifugal	
Design pressure	3,185 psig	
Design temperature	200° F	
Fluid	Reactor coolant	
Material	Stainless steel	
B.A. Transfer Pumps		
Number of units	2	
Type	Horizontal centrifugal	
Design flow	130 gpm	
Design pressure	200 psig	
Design temperature	200 °F	
Fluid	Boric acid water (approximately 7,000 ppmB)	
Material	Stainless steel	
B.A. Evaporator Feed Pumps		
Number of units	2	
Type	Horizontal centrifugal	
Design flow (process operation)	45 gpm	
Design flow (circulation operation)	130 gpm	
Design pressure	200 psig	
Design temperature	200° F	
Fluid	Reactor coolant	
Material	Stainless steel	
Regenerative Heat Exchanger		
Number of units	1	
Heat Transfer rate	27.4×10^6 BTU/h	
Type	Shell and tube type	
	Shell Side (Letdown)	Tube Side (Charging)
Design pressure	2485 psig	3185 psig
Design temperature	650 ° F	650 ° F
Design Flow rate	8.95×10^4 lb/h	7.98×10^4 lb/h
Design Inlet temperature	552.6° F	130.0° F
Design Outlet temperature	271.0° F	464.0° F
Material	Stainless steel	Stainless steel

Table 9.3.4-3 Chemical and Volume Control System Equipment Design Parameters (Sheet 2 of 6)

Letdown Heat Exchanger		
Number of unit	1	
Type	Single-shell pass U-tube	
Heat exchanger rate	24.2 x 10^6 BTU/H	
	Shell Side (CCW)	Tube side (Reactor coolant)
Design pressure	200 psig	700 psig
Design Temperature	300 ° F	400 ° F
Design flow rate	6.5 x 10^5 lb/h	8.95 x 10^4 lb/h
Design Inlet temperature	100° F	380° F
Design Outlet temperature	137.7° F	115° F
Material	Carbon steel	Stainless steel
Excess Letdown Heat Exchanger		
Number of unit	1	
Type	Vertical U-bend tube type	
Heat transfer rate	5.1~~2~~1 x 10^6 BTU/h	
	Shell side	Tube Side
Design pressure	200 psig	2485 psig
Design Temperature	300° F	650° F
Design Flow rate	1.37 x 10^5 lb/h	1.24 x 10^4 lb/h
Inlet temperature	100 ° F	55~~0~~2.~~9~~6 ° F
Outlet temperature	137.~~8~~4 ° F	165.0 ° F
Material	Carbon steel	Stainless steel
Seal Water Heat Exchanger		
Number of unit	1	
Type	Horizontal U-bend tube type	
Heat transfer rate	1.77 x 10^6 BTU/h	
	Shell Side	Tube Side
Design pressure	200 psig	150 psig
Design temperature	200° F	200° F
Design flow rate	1.25 x 10^5 lb/h	5.6 x 10^4 lb/h
Inlet temperature	100 ° F	146.7 ° F
Outlet temperature	113.5 ° F	115 ° F
Material	Carbon steel	Stainless steel
Mixed Bed Demineralizer		
Number of units	2	
Type	Vertical cylindrical	
Resin volume	70 ft^3	
Vessel capacity	100 ft^3	
Design pressure	300 psig	
Design temperature	150° F	
Design flow	180 gpm	
Material	Stainless steel	

ACRONYMS AND ABBREVIATIONS

A/B	auxiliary building
AAC	alternate alternating current
ac	alternating current
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BTP	branch technical position
CCWS	component cooling water system
CFR	Code of Federal Regulations
CGS	compressed gas supply system
COL	Combined License
CRDM	control rod drive mechanism
CS/RHRS	containment spray/residual heat removal system
CVCS	chemical and volume control system
CWS	circulating water system
dc	direct current
DCD	Design Control Document
DWS	demineralized water system
EAB	exclusion area boundary
ECCS	emergency core cooling system
EIA	Energy Information Administration
EPRI	Electric Power Research Institute
ESF	engineered safety features
ESW	essential service water
ESWS	Essential Service Water System
FCC	Federal Communications Commission
FMEA	failure mode and effects analysis
FOS	fuel oil storage and transfer system
FSAR	Final Safety Analysis Report
~~FTS~~	~~Fuel Transfer System~~
GDC	General Design Criteria
GTG	gas turbine generator
GWMS	gaseous waste management system
HEPA	high-efficiency particulate air

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

This section describes the heating, ventilation and air conditioning (HVAC) systems serving the plant during normal and ~~emergency~~abnormal conditions including SBO. HVAC systems are designed to provide suitable environment for plant equipment and personnel. Ventilation zones, air distribution and airflows migration are configured and arranged so that the ventilation air is drawn from the clean areas to areas of potentially greater radioactive contamination to a final filtration and exhaust systems discharging to the plant vent stack.

The HVAC systems airflow diagrams are shown on Figures 9.4.1-1 through 9.4.6-1. The area temperature and relative humidity during the plant normal and ~~emergency~~abnormal condition, including accident condition and LOOP condition, are described in Table 9.4-1.

The following are the reference sections where the various HVAC and related systems are covered:

Title	Section
Chilled Water System	9.2.7
Main Control Room HVAC System	9.4.1
Spent Fuel Pool Area Ventilation System	9.4.2
Auxiliary Building Ventilation System	9.4.3
Turbine Building Area Ventilation System	9.4.4
Engineered Safety Feature Ventilation System	9.4.5
Containment Ventilation System	9.4.6

The Main Control Room Heating, Ventilation and Air Conditioning System is subjected to the design objectives of RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" as it contains airborne radioactive material. A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in DCD Section 12.3.1. System and component design features addressing RG 4.21 (Ref.9.4.8-27) are summarized in Table 12.3-8. RG 4.21 is also applicable to the Auxiliary Building Ventilation System and the Engineered Safety Feature Ventilation System.

9.4.1 Main Control Room Heating, Ventilation and Air Conditioning System

The MCR HVAC System is designed to provide and control the proper environment in the MCR and other areas within the control room envelope (CRE) as defined in Chapter 6, Section 6.4. The MCR HVAC system complies with:

- 10 CFR 50, Appendix A, GDC 2,3,4,19
- 10 CFR 50.63
- RGs, 1.29, 1.52, 1.78, 1.155, 1.196, 1.197, and 4.21

The main steam/feedwater piping area HVAC system is designed to satisfy the following design bases:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) suitable to support the operation and assure the reliability of the electrical and mechanical components.
- Provide accessibility to system components for adjustment, maintenance and periodic inspection and testing of the system equipment and components to assure proper equipment function and reliability and system availability.

9.4.3.1.2.4 Technical Support Center (TSC) HVAC System

The TSC HVAC system is designed to satisfy the following design bases:

- Exclude entry of airborne radioactivity into the TSC envelope and remove radioactive material from the TSC envelope environment such that radiation doses to personnel are within the requirements of GDC 19 (10 CFR 50, Appendix A).
- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to assure personnel comfort and to support the operation of the control and instrumentation equipment and components.
- Support and maintain TSC habitability and permit personnel occupancy following plant emergency conditions.
- Provide accessibility to system components for adjustment, maintenance and periodic inspection and testing of the system components to assure proper equipment function and reliability and system availability.
- The TSC emergency filtration unit is designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.140 (Ref. 9.4.8-15).

9.4.3.2 System Description

9.4.3.2.1 Auxiliary Building HVAC System

The auxiliary building HVAC system is shown in Figure 9.4.3-1 and equipment and design data for the system are presented in Table 9.4.3-1. The COL Applicant is to determine the capacity of cooling and heating coils that are affected by site specific conditions. The auxiliary building HVAC system does not serve any safety function, with the exception of the safety-related isolation dampers such as penetration area supply and exhaust line isolation dampers, safeguard component area supply and exhaust isolation dampers and auxiliary building exhaust line isolation dampers. Therefore, the auxiliary building HVAC system is not safety-related. Non-safety related equipment and ductwork within areas containing safety-related equipment are supported as seismic

- Indication of outlet airflow of air handling units and low airflow alarm.
- Indication of differential pressure across a filter bank in the air handling units.
- Indication of air handling unit inlet and outlet air temperature.
- Alarm on air handling unit electric heating coil outlet temperature.

9.4.3.5.4 Technical Support Center (TSC) HVAC System

The TSC HVAC System is operable from MCR. The following instrumentation is available in the MCR.

- Indication of the status of air handling and emergency filtration units.
- Indication of the status of TSC air intake, toilet/kitchen exhaust line isolation dampers.
- Indication of the TSC envelope differential pressure.
- Indication of the TSC emergency filtration unit electric heating coil outlet temperature and high temperature alarm.
- Indication of the TSC emergency filtration unit charcoal adsorber outlet air temperature and high, high-high temperature alarm.
- TSC air handling unit electric heating coil outlet high temperature alarm.
- TSC high temperature alarm.
- TSC emergency filtration unit total differential pressure alarm.
- TSC emergency filtration unit HEAP filter differential pressure alarm.
- TSC emergency filtration unit outlet airflow rate low and high alarm.
- TSC air handling unit outlet airflow rate low alarm.
- Alarm on smoke detection.
- Alarm on airborne radioactivity detection at the outside air intake.

9.4.4 Turbine Building Area Ventilation System

The turbine building area ventilation system maintains a suitable environment for the operation of equipment in turbine building. This system includes the following:

- General mechanical areas ventilation system

- Electrical equipment areas heating, ventilation, and air conditioning (HVAC) system

9.4.4.1 Design Basis

9.4.4.1.1 Safety Design Bases

The turbine building area is not expected to include airborne radioactive contamination. Safety-related equipment is not located in this area. Therefore, the turbine building area ventilation system does not serve any safety-related function, and thus, has no safety design bases.

9.4.4.1.2 Non-safety Power Generation Design Bases

The turbine building area ventilation systems has following design bases:

- The general mechanical areas ventilation system is designed to maintain~~s~~ a suitable environment in the general mechanical areas during normal operating condition.
- In the event of the presence of smoke, the general mechanical areas ventilation system purges the smoke in the general mechanical areas.
- The electrical equipment areas HVAC system maintains a suitable environment in the electrical equipment areas during normal operating, loss of offsite power conditions.
- In the event of the presence of smoke, the electrical equipment areas HVAC system purges the smoke in electrical equipment areas.
- The electrical equipment areas HVAC system maintains the hydrogen concentration well below 1% by volume in battery room.
- The turbine building ventilation system is designed in accordance with ANSI/AMCA and ASHRAE.

Refer to the design temperature and relative humidity for the turbine building area in Table 9.4-1.

9.4.4.2 System Description

The turbine building area ventilation system is shown on Figure 9.4.4-1. Design data for the principal systems components are presented in Table 9.4.4-1.

9.4.4.2.1 General Mechanical Areas Ventilation System

The general mechanical areas ventilation system consists of turbine building roof ventilation fans, basement area supply fans, basement area exhaust/circulating fans, wall louvers and sampling room HVAC system. This system is once through using outdoor air for cooling.

The system is thermostatically controlled by area temperature controllers to start or stop the roof fans and open or close the wall louver dampers to maintain the design temperature limits. Within the turbine building are areas with exterior walls.

In the event of the presence of smoke, selected roof fans are actuated to purge the smoke. If a fire is detected in the Turbine building, all ~~27~~ roof fans shut down automatically. Once the fire has been extinguished, smoke purge operation is initiated manually by restarting fans as needed from the main control room.

A supply air louver is not installed in the basement area. Therefore, outdoor air is to be provided to this area by a basement area supply fans with associated distribution ductwork, to maintain the proper design temperature limits. This area has ~~a~~ local thermostats and temperature controllers to adjust airflow by controlling the fan in response to the area temperatures. ~~A b~~Basement area exhaust ~~/circulating fan is~~fans are provided between the basement and the first floor to keep air circulating and to exhaust hot air from the basement.

The sampling room is a stand-alone area in the general mechanical areas and room temperatures is ~~be~~ maintained by ~~split unit type~~ the sampling room HVAC system.

9.4.4.2.2 Electrical Equipment Areas HVAC System

The electrical equipment areas HVAC system consists of two 100% non-Class 1E electrical room air handling units and non-Class 1E battery rooms common exhaust system. This HVAC system is powered from the alternate ac power source and operated during LOOP condition.

This HVAC system serves ~~to~~ the electrical equipment area. This area is divided into two floors(1FL and 2FL). Each floor consists of a non-Class 1E battery room and a non-Class 1E electrical room. Electrical room includes a permanent bus backed up by the alternate ac power source.

Each floor is served by an electrical equipment air handling unit. The air handling unit consists of an air intake low efficiency pre-filter, high efficiency final filter, electric heating coil, chilled water cooling coil, supply fan, return air fan, and associated controls. The cooling coil of each air handling unit is supplied with chilled water from the non-essential chilled water system (Subsection 9.2.7.2). The air handling unit automatically uses outside air or chilled water cooling coil or the electric heating coil to maintain room temperatures within the design temperature limits.

The battery rooms common exhaust system has two 100% exhaust fans, with one in standby. When one fan fails, the fan failure is alarmed in the main control room and the other one starts automatically. This system maintains the hydrogen concentration well below 1% by volume in both battery rooms.

Smoke detectors located in each floor detect the presence of smoke and automatically shutdown the air handling unit and alarm in the main control room. As soon as the source of smoke is determined to be from outside or a fire inside the room and the fire has been extinguished, the system may be manually placed into the smoke purge mode of operation from the control room. The chilled water cooling coil of the air handling units is automatically positioned for full chilled water flow to avoid the possibility of freezing during low outside ambient temperatures. The exhaust fans for the battery rooms will not trip on smoke detection.

All duct penetrations in the fire walls are protected by fire dampers to prevent the spread of fire from an affected area to the adjacent redundant component areas.

9.4.4.3 Safety Evaluation

The turbine building area ventilation system does~~se~~ not serve any safety-related function, and thus, requires no safety evaluation.

9.4.4.4 Inspection and Testing Requirements

Air handling equipment is factory tested in accordance with Air Movement and Control Association Standard. Air filters are tested in accordance with American Society of Heating, Refrigerating, and Air Conditioning Engineers Standard. Cooling coils are tested in accordance with Air Conditioning and Refrigeration Institute Standard.

- Each component in the turbine building area ventilation system is provided with proper access for initial and periodic testing and inspection during normal operation.
- Each system and component is operated and adjusted to design operating conditions during the plant preoperational test program.
- System airflows are to be balanced to obtain design airflows that will maintain the design temperature limits throughout the served areas.
- Air handling equipment is factory tested in accordance with Air Movement and Control Association Standard (Ref. 9.4.8-16, Ref. 9.4.8-17, Ref. 9.4.8-18). Air filters are tested in accordance with American Society of Heating, Refrigerating, and Air Conditioning Engineers Standard (Ref. 9.4.8-19, Ref. 9.4.8-20). Cooling coils are tested in accordance with Air Conditioning and Refrigeration Institute Standard (Ref. 9.4.8-21, Ref. 9.4.8-22).
- System instruments and automatic controls are to be calibrated to insure proper set points and confirm proper sequence of operation at all system operating modes.
- The system is operated and tested initially with regard to flow paths, flow capacity and component operability.

- 10 CFR 50, Appendix A, GDC 17 is satisfied in part for the essential electrical components of the ESF Ventilation System, such as contacts and relays. This is accomplished by protecting the components from accumulated dust and particulate materials by enclosing the components in dust-tight cabinets and taking outdoor air through air filters from a height of at least 7 meters (20 feet) above ground level.

9.4.5.1.1.1 Annulus Emergency Exhaust System

During normal plant operation, the penetration areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident, the safety-related isolation dampers automatically isolate the supply and exhaust line of the auxiliary building HVAC system. The annulus emergency exhaust system is designed to satisfy the following design basis.

- The emergency exhaust filtration units are designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.52 (Ref. 9.4.8-3).

- The system is designed to mitigate the consequences of postulated accidents by removing the airborne radioactive material that may leak from containment.

- The system remains functional during and after a design basis accident.

- The system maintains a negative pressure in the penetration and safeguard comportment areas relative to the adjacent areas (Chapter 6, Section 6.5.1).

9.4.5.1.1.2 Class 1E Electrical Room HVAC System

The Class 1E electrical room HVAC system is designed to satisfy the following design basis.

- Maintain proper operating environmental conditions within the Class 1E electrical rooms (Table 9.4-1) during normal and design basis accident.

~~•Maintain the hydrogen concentration below 1% by volume of Class 1E battery room.~~

9.4.5.1.1.3 Safeguard Component Area HVAC system

During normal plant operation, the safeguard component areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the safety-related redundant isolation dampers automatically isolate the supply and exhaust line of the auxiliary building HVAC system. The safeguard component area HVAC system is designed to satisfy the following design basis:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to support the operation of the control and instrumentation equipment and components in the individual safeguard component areas during a design basis accident or LOOP.

9.4.5.1.1.4 Emergency Feedwater Pump Area HVAC System

During normal plant operation, the emergency feedwater pump (motor-driven) areas are served by the auxiliary building HVAC system (Section 9.4.3), while the emergency feedwater pump (turbine-driven) areas are served by the emergency feedwater pump (turbine-driven) area air handling unit. The emergency feedwater pump area HVAC system is designed to satisfy the following design basis:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to support the operation of the control and instrumentation equipment and components in the individual emergency feedwater pump areas during a design basis accident or LOOP.

9.4.5.1.1.5 Safety Related Component Area HVAC System

During normal plant operation, the safety-related component areas are served by the auxiliary building HVAC system (Section 9.4.3). The safety-related component area HVAC system is designed to satisfy the following design basis:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to support the operation of the control and instrumentation equipment and components in the individual safety related component areas during a design basis accident or LOOP.

9.4.5.1.2 Power Generation Design Basis

The ESF ventilation system is designed to satisfy the following design basis:

- Provide accessibility for adjustment and periodic inspection, maintenance and testing of the system equipment and components.

The Class 1E electrical room HVAC system stops within one hour after SBO occurs until alternate ac gas turbine generator restores power. However, all Class 1E cabinets are designed to keep their integrity during loss of a HVAC system (Chapter 8, Section 8.4).

The Class 1E electrical room HVAC system is designed to maintain the hydrogen concentration below 1% by volume of Class 1E battery room.

9.4.5.2 System Description

9.4.5.2.1 Annulus Emergency Exhaust System

The annulus emergency exhaust system is an ESF system designed for fission product removal and retention. The system is shown in Figure 9.4.5-1 and the system equipment design data is presented in Table 9.4.5-1. The annulus emergency exhaust system consists of two redundant divisions, each sized to satisfy 100% capacity.

filter, a high efficiency filter, an electric heating coil, a chilled water cooling coil, a supply fan, and associated controls. The cooling coil of each system's air handling unit is supplied with chilled water from the corresponding essential chilled water system (Section 9.2.7). Return air from the electrical room is drawn through the return air ductwork by the system's return air fans. Both air handling units are connected to a common air distribution duct through their discharge air isolation dampers.

Train pair A&B and train pair C&D, each is connected to a single air distribution system. The air distribution system is qualified in accordance with seismic category I requirements. Conditioned air is distributed to the following areas:

- Class 1E instrumentation and control (I&C) rooms
- Class 1E electrical rooms
- Class 1E uninterruptible power supply (UPS) rooms
- Class 1E Battery and battery charger rooms
- MCR/Class 1E electrical HVAC equipment rooms
- Remote shutdown console room
- Control rod drive mechanism (CRDM) cabinet room (non-safety)
- M-G set and M-G set panel rooms (non-safety)
- Leakage rate testing (LRT) room (non-safety)
- Reactor trip breaker room
- AA~~C~~C selector circuit panel room

The return air from these areas is drawn by the corresponding HVAC train through the seismic category I ductwork.

The volume of the air exhausted from battery rooms by the corresponding battery exhaust fans is sufficient to maintain the hydrogen concentration well below 1% by volume of battery room.

~~Rooms with high heat loss during the cold season are provided with non safety-related unit heaters or in-duct electric heaters in their supply air branches.~~ The safety-related in-duct heaters are provided in supply air branches to Remote Shutdown Console Room, Class 1E Battery Rooms, Class 1E I&C Rooms and Class 1E Electrical Room & MCR HVAC Equipment Rooms. These electric heaters are classified as equipment class ~~5~~3 and seismic category ~~I~~I.

9.4.5.2.2.3 Emergency Operation Mode

Upon receipt of the ECCS actuation signal, the Class 1E electrical room HVAC system automatically switches to emergency operation by initiating the following control functions:

- The operating trains continue to run and the standby trains start.
- The operating battery exhaust fans continue and the standby fans start.
- Following automatic initiation of emergency operation, two of the HVAC trains and two of the battery exhaust fans may be manually de-energized and placed on standby status.

9.4.5.2.3 Safeguard Component Area HVAC system

During normal plant operation, safeguard component areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the safeguard component areas are cooled by individual safeguard component area air handling units. The safeguard component area includes the CS/RHR pump rooms, SI pump rooms, CS/RHR heart exchanger rooms, AHU rooms, R/B sump tank rooms.

A rise of the safeguard component area temperature reaching the setpoint of the switch is to a cause the associated fan to start and the air handling unit inlet damper and outlet damper open upon receipt of their respective fan run signals. Reverse operation occurs upon a temperature decrease below the setpoint of the switch.

Each air handling unit consists of, in the direction of airflow, an electric heating coil, a cooling coil, a supply fan and associated controls. The safeguard component area HVAC system is shown in Figure 9.4.5-3 and the equipment design data is presented in Table 9.4.5-1. The COL Applicant is to determine the capacity of heating coils that are affected by site specific conditions. The cooling coils are supplied with chilled water from the essential chilled water system (Section 9.2.7).

Upon safeguard component area high temperature, the chilled water cooling coil control valve for the corresponding air handling units is automatically positioned for full chilled water flow to prevent the temperature rise.

Upon electric heating coil outlet high temperature, the electric heating coil is automatically tripped to prevent the abnormal heating.

The air handling unit trains A, B, C and D provide 100% of the heating and cooling requirements of their associated equipment room.

The function of the backdraft dampers at the common duct section that interfaces between the annulus emergency exhaust system and the auxiliary building HVAC system is described in Subsection 9.4.5.2.1.

9.4.5.2.4 Emergency Feedwater Pump Area HVAC System

9.4.6.1.2.4 Containment Purge System

The containment purge system, with the exception of the safety-related and seismic category I containment isolation valves, is designed to satisfy the following design bases:

- Maintain low concentrations of radioactivity in the containment atmosphere to allow access during maintenance and inspection activities.
- Provide relief from pressure build-up caused by instrument air leakage and containment temperature fluctuations.
- The supply air to the containment is dehumidified and tempered to minimize the condensation on the containment ventilation system's cooling coils and supply air duct inside the containment.
- The containment low volume purge exhaust filtration units and containment high volume purge exhaust filtration unit are designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.140 (Ref. 9.4.8-15).

9.4.6.2 System Description

9.4.6.2.1 Containment Fan Cooler System

The containment fan cooler system is shown in Figure 9.4.6-1 and the equipment and design data is presented in Table 9.4.6-1. The containment fan cooler system does not serve any safety function. Therefore, it is not safety-related. Non-safety related equipment and ductwork, including supports, in~~within~~ areas containing safety-related equipment are ~~supported as~~ seismic Category II to prevent adverse interaction with safety-related systems during a seismic event.

The system consists of four fan cooler units, each sized for 1/3 of the total containment heat load, dampers, ductwork and associated instrumentation and controls. During normal operation, three units are required to operate while the other unit remains on standby. Each fan cooler unit consists of a cooling coil and an axial fan. There are backdraft dampers located on the discharge ductwork to the header compartment.

The containment air is cooled by the operating containment fan coolers. The cooling coils are supplied with chilled water from the non-essential chilled water system. Air is distributed inside the containment through the header compartment and the distribution ductwork system. The cooling airflow that is delivered to each SG compartment and pressurizer compartment through the header compartment to maintain each compartment in proper temperature is 19,000 cfm and 13,500~~0~~ cfm, respectively.

The chilled water control valve of each unit is controlled by an area temperature controller that modulates the chilled water flow to maintain the average containment air temperature below 120° F (Table 9.4-1).

Table 9.4-1 Area Design Temperature and Relative Humidity (Sheet 4 of 4)

Notes

Note1: Outside air ambient design temperature condition is as follows:

(a) 0% exceedance dry bulb and wet bulb temperature of site ambient temperature condition (See Chapter 2)

(b) 1% exceedance dry bulb and wet bulb temperature of site ambient temperature condition (See Chapter 2)

(c) -5^{o}F (minimum) to 95^{o}F dry bulb / 77^{o}F coincident wet bulb (maximum)

Note2: Location: PCCV, Prestressed concrete containment vessel; RB, Reactor building; AB, Auxiliary building; ACB, Access building; PSB, Power source building; TB, Turbine building.

Note3: Smoke purge mode is not required the temperature and humidity condition.

Note4: The Containment High Volume Purge System maintains proper environmental condition at the design temperature range during refueling condition. The Containment Low Volume Purge System is not mean to be used for containment cooling and heating (See Subsection 9.4.6.2.4.1.).

Note5: During LOOP condition only.

Note6: During the gas turbine generator stop condition

Table 9.4.3-1 Equipment Design Data (Sheet 1 of 2)

Auxiliary Building Air Handling Unit	
Number of Units	2
Equipment Class	~~5~~9
Seismic Category	Non-Seismic
Unit Airflow Capacity, cfm	98,000
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	10-35%
Medium Efficiency Filter Efficiency	45-75%
Cooling Coil Type	Chilled Water
Heating Coil Type	Steam
Auxiliary Building Exhaust Fan	
Number of Fans	3
Equipment Class	~~5~~8
Seismic Category	Non-Seismic
Fan Airflow, cfm	108,000
Fan Type	Centrifugal
Non-Class 1E Electrical Room Air Handling Unit	
Number of Units	2
Equipment Class	~~5~~9
Seismic Category	Non-Seismic
Unit Airflow Capacity, cfm	40,000
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	10-35%
High Efficiency Filter Efficiency	85-95%
Cooling Coil Type	Chilled Water
Heating Coil Type	Steam
Non-Class 1E Electrical Room Return Air Fan	
Number of Units	2
Equipment Class	~~5~~9
Seismic Category	Non-Seismic
Fan Airflow Capacity, cfm	36,250
Fan Type	Vane Axial
Non-Class 1E Battery Room Exhaust Fan	
Number of Fans	2
Equipment Class	5
Seismic Category	Non-Seismic
Fan Airflow Capacity, cfm	7,500
Fan Type	Vane Axial

Table 9.4.3-1 Equipment Design Data (Sheet 2 of 2)

Main Steam / Feedwater Piping Areas Air Handling Unit	
Number of Units	4
Equipment Class	~~5~~9
Seismic Category	Non-Seismic
Unit Airflow Capacity, cfm	5,500
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	10-35%
Cooling Coil Type	Chilled Water
Heating Coil Type	Electric
Technical Support Center Air Handling Unit	
Number of Units	1
Equipment Class	5
Seismic Category	Non-Seismic
Unit Airflow Capacity, cfm	17,000
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	10-35%
High Efficiency Filter Efficiency	85-95%
Cooling Coil Type	Chilled Water
Heating Coil Type	Electric
Technical Support Center Emergency Filtration Unit	
Number of Units	1
Equipment Class	5
Seismic Category	Non-Seismic
Unit Airflow Capacity, cfm	1,800
Unit Fan Type	Centrifugal
After-Filters Type	High Efficiency
Adsorber Type	Impregnated charcoal
Charcoal Adsorber Radioiodine Efficiency	95% minimum
HEPA Filter Efficiency (design basis)	99%
HEPA Filter Efficiency (design specification)	99.97%, 0.30 micron particles
High Efficiency Filter Efficiency	85-95%
Heating Coil Type	Electric
Heating Coil Capacity, kW	9
Technical Support Center Toilet/Kitchen Exhaust Fan	
Number of Units	1
Equipment Class	~~5~~9
Seismic Category	Non-Seismic
Fan Airflow Capacity, cfm	1,000
Fan Type	Vane Axial

Table 9.4.4-1 Equipment Design Data

Non-Class 1E Electrical Equipment Area Air Handling Unit	
Number of Units	2
Equipment Class	5
Seismic Category	Non-Seismic
~~Air flow capacity, cfm~~	~~27,000~~
Unit Fan Type	Centrifugal
~~Return Air Fan type~~	~~Centrifugal~~
Cooling coil type	Chilled Water
Heating coil type	Electric
Basement Area Supply Fan	
~~Number of Fans~~	~~1~~
Equipment Class	5
Seismic Category	Non-Seismic
Fan type	Variable pitch axial
~~Fan flow, cfm~~	~~50,000~~
Basement Area Exhaust~~/Circulating Supply~~ Fan	
~~Number of Fans~~	~~1~~
Equipment Class	5
Seismic Category	Non-Seismic
Fan type	Vane axial
~~Fan Airflow Capacity, cfm~~	~~50,000~~
Turbine Building Roof Ventilation Fan	
~~Number of Fans~~	~~27~~
Equipment Class	5
Seismic Category	Non-Seismic
Fan type	Power roof ventilator
~~Fan Airflow Capacity, cfm~~	~~100,000~~
Non-Class 1E Battery Room Exhaust Fan	
~~Number of Fans~~	~~2~~
Equipment Class	5
Seismic Category	Non-Seismic
Fan type	Centrifugal
~~Fan Airflow Capacity, cfm~~	~~2,000~~

Table 9.4.5-1 Equipment Design Data (Sheet 1 of 4)

Class 1E Electrical Room Air Handling Unit	
Number of Units	4
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	40,000 - train A, B 52,000 - train C, D
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	25-35%
High Efficiency Filter Efficiency	80-95%
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	1,650,000 - train A, B 2,2~~5~~90,000 - train C, D
Heating Coil Type	Electric
Class 1E Electrical Room Return Air Fan	
Number of Fans	4
Equipment Class	3
Seismic Category	I
Fan Airflow Capacity, cfm	37,400 - train A, B 49,400 - train C, D
Fan Type	Axial
Class 1E Battery Room Exhaust Fan	
Number of Fans	4
Equipment Class	3
Seismic Category	I
Fan Airflow Capacity, cfm	2,600
Fan Type	Axial
Annulus Emergency Exhaust Filtration Unit	
Number of Units	2
Equipment Class	2
Seismic Category	I
Unit Airflow Capacity, cfm	5,600
Unit Fan Type	Centrifugal
HEPA Filter Efficiency	99.97%, 0.30 micron particles
High Efficiency Filter Efficiency	80-95%

Table 9.4.5-1 Equipment Design Data (Sheet 2 of 4)

Safeguard Component Area Air Handling Unit	
Number of Units	4
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	5,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	180,000
Heating Coil Type	Electric
Emergency Feedwater Pump (M/D) Area Air Handling Unit	
Number of Units	2
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	2,100
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	110,000
Heating Coil Type	Electric
Emergency Feedwater Pump (T/D) Area Air Handling Unit	
Number of Units	2
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	1,300
Unit Fan Type	Centrifugal
Low Efficiency Filter Efficiency	25-35%
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	~~60,000~~62,000
Heating Coil Type	Electric
Penetration Area Air Handling Unit	
Number of Units	4
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	5,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	330,000
Heating Coil Type	Electric

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet 1 of ~~6~~7)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
1. Annulus Emergency Exhaust System					
1.1	Annulus emergency exhaust filtration unit fan	Draw the air from penetration and safeguard component area	Fails to start upon the demand signal Trip for any reason	None: One 100% capacity Annulus emergency exhaust filtration unit train remains available; Minimum one is required.	Low airflow alarm, fan motor current, and RUN indication in the MCR
1.2	Annuls emergency exhaust damper VRS-EHD-001A (-001B analogous)	Open to provide flow path	Failure to open upon the demand signal	None: One 100% capacity Annulus emergency exhaust filtration unit train remains available; Minimum one is required.	Damper position indication in the MCR
1.3	Safeguard component area exhaust damper VRS-EHD-002A (-002B analogous)	Open to provide flow path	Failure to open upon the demand signal	None: One 100% capacity Annulus emergency exhaust filtration unit train remains available; Minimum one is required.	Damper position indication in the MCR
1.4	Annulus emergency exhaust filtration unit outlet damper VRS-EHD-003A (-003B analogous)	Open to provide flow path	Failure to open upon the demand signal	None: One 100% capacity Annulus emergency exhaust filtration unit train remains available; Minimum one is required.	Low airflow alarm and damper position indication in the MCR

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet 2 of 67)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
2. Class 1E Electrical Room HVAC System					
2.1	Class 1E electrical room air handling unit	Deliver to conditioned air to Class 1E electrical rooms	Fails to start upon the demand signal Trip for any reason	None: Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.	Low airflow alarm, high temperature alarm, fan motor current, and RUN indication in the MCR
2.2	Class 1E electrical room return air fan	Draw the air from Class 1E electrical rooms	Fails to start upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three Class 1E electrical room return air fan trains remain available; Minimum two are required.	Fan motor current, and RUN indication in the MCR
2.3	Class 1E battery room exhaust fan	Exhaust the air from Class 1E battery rooms	Fails to start upon the demand signal Trip for any reason	None: Three Class 1E battery room exhaust fans remain available; Minimum two are required.	Low airflow alarm, fan motor current, RUN indication, and damper position indication in the MCR
2.4	Class 1E electrical room outside air intake isolation damper VRS-EHD-201A (-201B, -201C, and -201D analogous)	Open to provide flow path	Failure to open upon the demand signal	None: Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.	Damper position indication in the MCR

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet 3 of ~~6~~7)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
2.5	Class 1E electrical room return air fan inlet damper VRS-EHD-203A (-203B, -203C, and -203D analogous)	Open to provide flow path	Failure to open upon the demand signal	None: Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.	Low airflow alarm and damper position indication in the MCR
~~2.6~~	~~Class 1E electrical room return air fan inlet damper VRS-EHD-203A (-203B, -203C, and -203D analogous)~~	~~Open to provide flow path~~	~~Failure to open upon the demand signal~~	~~None:~~ ~~Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.~~	~~Low airflow alarm and damper position indication in the MCR~~
2.6~~7~~	Class 1E electrical room air handling unit inlet damper VRS-EHD-204A (-204B, -204C, and -204D analogous)	Open to provide flow path	Failure to open upon the demand signal	None: Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.	Low airflow alarm, fan motor current, RUN indication, and damper position indication in the MCR
2.7~~8~~	Class 1E electrical room exhaust line isolation damper VRS-AOD-205A (-205B,-205C, and -205D analogous)	Close to isolate from the outside air	Failure to close upon the demand signal	None: The affected train is isolated by the Class 1E electrical room air handling unit outlet and return air fan inlet dampers.	Damper position indication in the MCR

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet 4 of ~~6~~7)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
2.8~~9~~	Class 1E battery room exhaust fan inlet damper VRS-EHD-251A (-251B, -251C, and -251D analogous)	Open to provide flow path	Failure to open upon the demand signal	None: Three Class 1E battery room exhaust fans remain available; Minimum two are required.	Low airflow alarm, and damper position indication in the MCR
2.9~~10~~	Class 1E battery room exhaust fan outlet~~inlet~~ damper VRS-EHD-252A (-252B, -252C, and -252D analogous)	Open to provide flow path	Failure to open upon the demand signal	None: Three Class 1E battery room exhaust fans remain available; Minimum two are required.	Low airflow alarm, and damper position indication in the MCR
2.10	In-duct heater	Maintain room temperature in rooms with high heat loss during the cold season.	Fails to heat the supply air upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three Class 1E electrical room air handing unit trains remain available; Minimum two are required.	Temperature indication

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet ~~4~~5 of ~~6~~7)

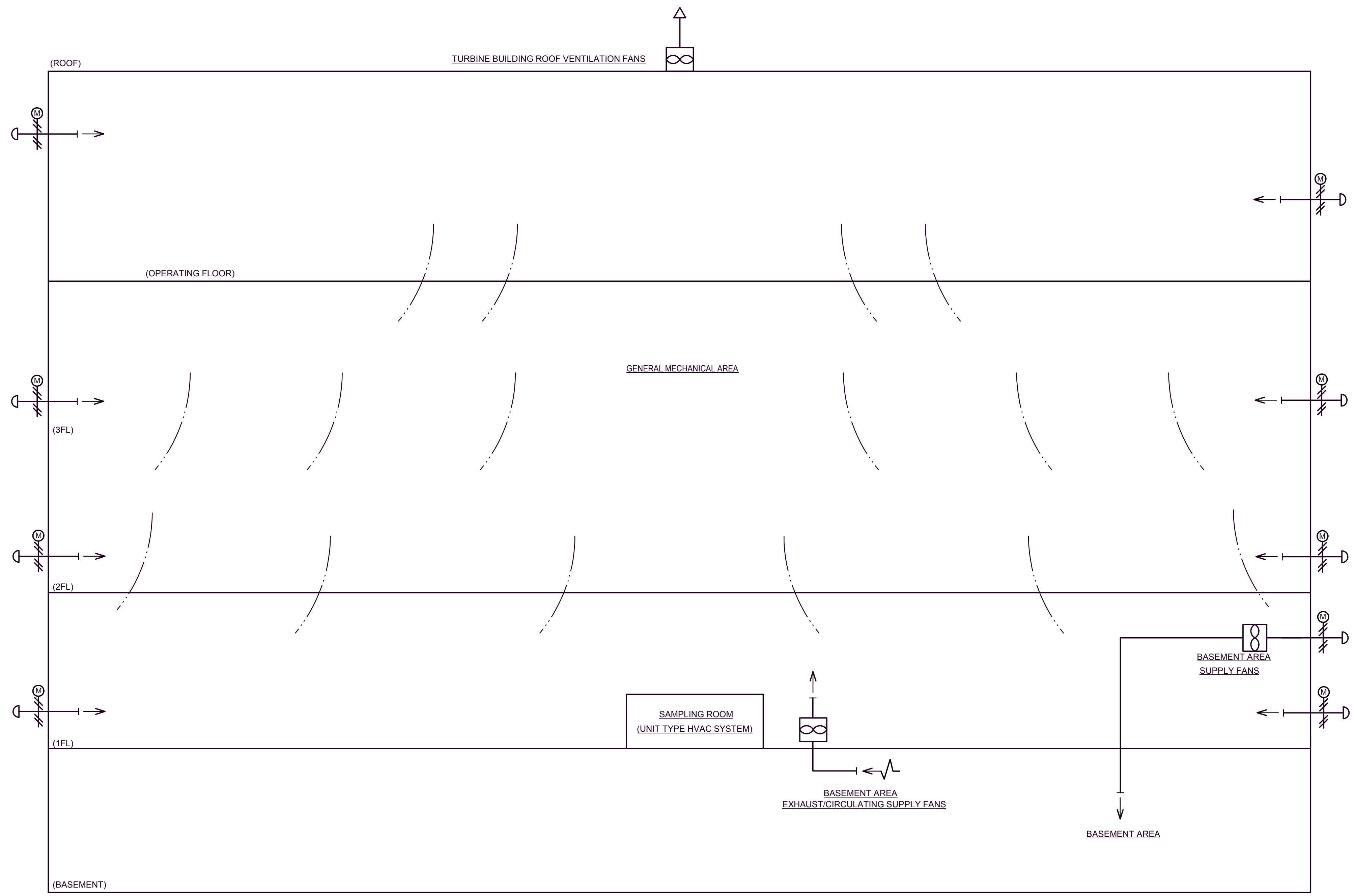
Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
3. Safeguard Component Area HVAC System					
3.1	Safeguard component area air handling unit	Deliver to conditioned air to safeguard component area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	Low airflow alarm, high temperature alarm, fan motor current, and RUN indication in the MCR
3.2	Safeguard component area air handling unit inlet damper VRS-MOD-301A (-301B, -301C, and -301D analogous)	Open to provide airflow path	Failure to open on demand	None: Three air handling unit trains remain available; Minimum two are required.	Low airflow alarm and damper position indication in the MCR
3.3	Safeguard component area air handling unit outlet damper VRS-MOD-302A (-302B, -302C, and -302D analogous)	Open to provide flow path	Failure to open on demand	None: Three air handling unit trains remain available; Minimum two are required.	Low airflow alarm and damper position indication in the MCR

Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet ~~5~~6 of ~~6~~7)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
4. Emergency Feedwater Pump Area HVAC System					
4.1	Emergency Feedwater pump area air handling unit	Deliver to conditioned air to emergency feedwater pump area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	High temperature alarm, fan motor current, and RUN indication in the MCR
5. Safety Related Component Area HVAC System					
5.1	Penetration area air handling unit	Deliver to conditioned air to penetration area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	High temperature alarm, fan motor current, and RUN indication in the MCR
5.2	Annulus emergency filtration unit area air handling unit	Deliver to conditioned air to annulus emergency filtration unit area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: One air handling unit remains available; Minimum one is required.	High temperature alarm, fan motor current, and RUN indication in the MCR

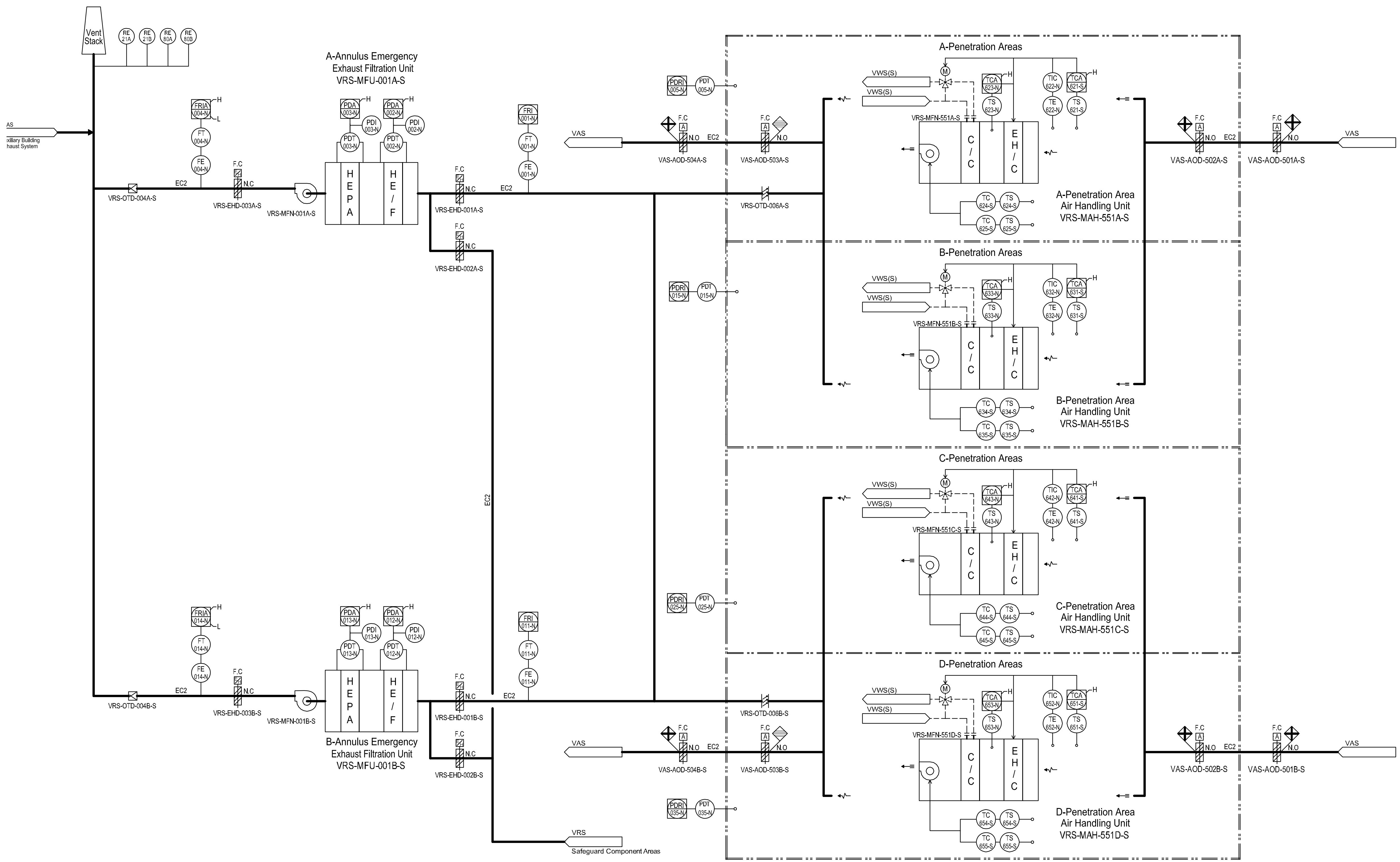
Table 9.4.5-2 Engineered Safety Features Ventilation Systems Failure Modes and Effects Analysis (Sheet ~~6~~7 of ~~6~~7)

Item	Component	Safety Function	Failure Mode	Effect on System Operation	Failure Detection Method
5.3	Charging pump area air handling unit	Deliver to conditioned air to charging pump area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: One air handling unit remains available; Minimum one is required.	High temperature alarm, fan motor current, and RUN indication in the MCR
5.4	CCW pump area air handling unit	Deliver to conditioned air to CCW pump area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	High temperature alarm, fan motor current, and RUN indication in the MCR
5.5	Essential chiller unit area air handling unit	Deliver to conditioned air to essential chiller unit area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	High temperature alarm, fan motor current, and RUN indication in the MCR
5.6	Spent fuel pit pump area air handling unit	Deliver to conditioned air to spent fuel pit pump area	Fails to start to upon the demand signal Trip for any reason Loss of room temperature control upon demand signal	None: Three air handling units remain available; Minimum two are required.	High temperature alarm, fan motor current, and RUN indication in the MCR



NOTE:
1. ALL PIPING,VALVES AND COMPONENTS IN THIS SHEET ARE CLASSIFIED AS EQUIPMENT CLASS 9.

Figure 9.4.4-1 Turbine Building Area Ventilation System Flow Diagram (Sheet 1 of 2)



REMARK:
PLANT DESIGNATION OF EQUIPMENT AND VALVE NUMBERS ARE OMITTED IN THIS DRAWING.
~~STD~~ xxx - ### - +++

Figure 9.4.5-1 Annulus Emergency Exhaust System Flow Diagram

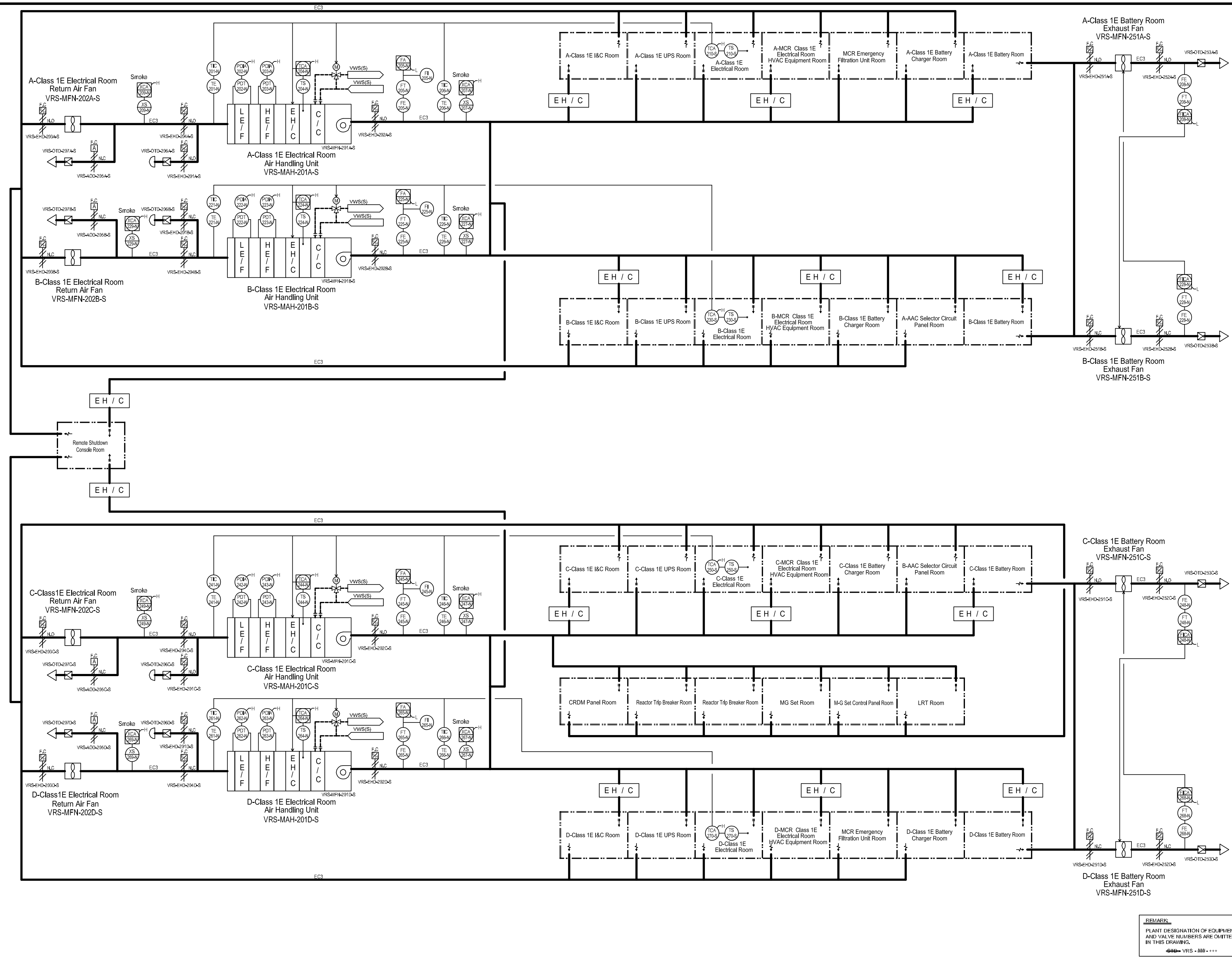


Figure 9.4.5-2 Class 1E Electrical Room HVAC System Flow Diagram

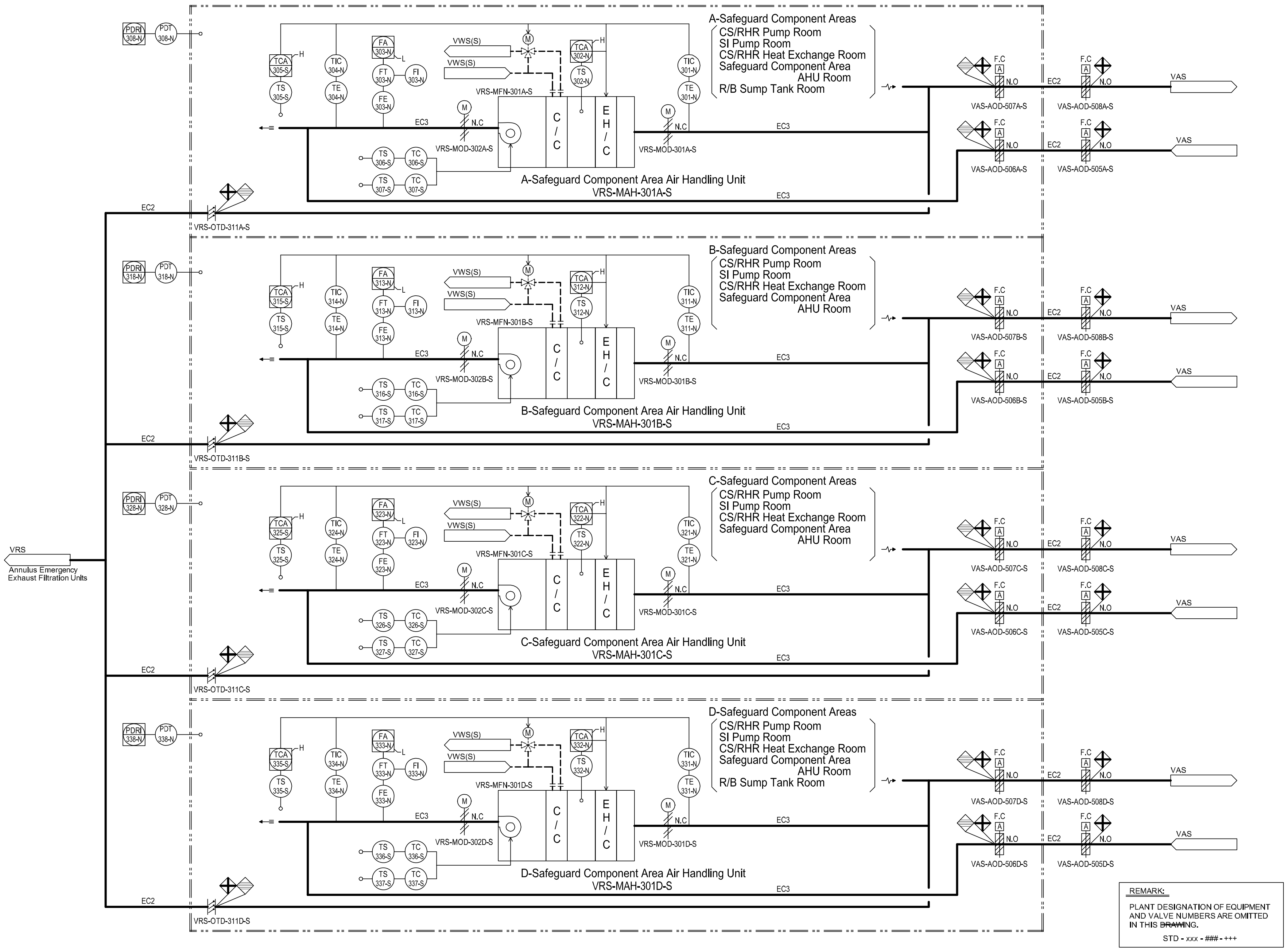


Figure 9.4.5-3 Safeguard Component Area HVAC System Flow Diagram

ACRONYMS AND ABBREVIATIONS

A/B	auxiliary building
ac	alternating current
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BTP	branch technical position
CCWS	component cooling water system
CFR	Code of Federal Regulations
CGS	compressed gas supply system
COL	Combined License
CRDM	control rod drive mechanism
CS/RHRS	containment spray/residual heat removal system
CVCS	chemical and volume control system
CWS	circulating water system
dc	direct current
DCD	Design Control Document
DWS	demineralized water system
ECCS	emergency core cooling system
EIA	Energy Information Administration
EPRI	Electric Power Research Institute
ESF	engineered safety features
ESW	essential service water
ESWS	Essential Service Water System
FCC	Federal Communications Commission
FMEA	failure mode and effects analysis
FOS	fuel oil storage and transfer system
FSAR	Final Safety Analysis Report
~~FTS~~	~~Fuel Transfer System~~
GDC	General Design Criteria
GTG	gas turbine generator
GWMS	gaseous waste management system
HEPA	high-efficiency particulate air
HID	high intensity discharge
I&C	instrumentation and control

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

The primary objectives of the US-APWR fire protection program are: to minimize the potential for fire and explosions to occur; to rapidly detect, control, and extinguish any fire that may occur; and to assure that any fire that may occur will not prevent the performance of necessary safe-shutdown functions and will not significantly increase the risk of radioactive releases to the environment. In addition, the US-APWR fire protection systems are designed such that any system failure or inadvertent operation does~~dose~~ not adversely impact the ability of the structures, systems and components (SSCs) important to safety to perform their safety functions. The US-APWR fire protection program primarily consists of the following elements:

- Comprehensive identification and analysis of fire and explosion hazards.
- Organization and staff positions responsible for management and implementation of the fire protection program.
- Fire prevention program consisting of administrative policy, procedures, and practices for training of general plant personnel; control of fire hazards; inspection, testing and maintenance of fire protection systems and features; interaction with plant design and modification program; control of fire system outages and impairments; and fire protection program quality assurance.
- Automatic fire detection, alarm, and suppression systems, including fire water supply and distribution systems.
- Manual suppression capability including portable fire extinguishers and standpipes, hydrants, hose stations, fire department connections, fire brigade organization, training, qualification, equipment, and drills; emergency plans and procedures; and offsite mutual aid capabilities.
- Building design for fire protection including layout of fire areas, fire barrier design and qualification testing, interior finish, electrical system design, ventilation system design, drainage systems, and other systems and features for minimizing the threat of fire.
- Post-fire safe-shutdown analysis and procedures that demonstrate that the plant can achieve and maintain safe-shutdown in the event of a fire.
- Probabilistic risk assessment (PRA) that identifies relative fire risks and vulnerabilities.

As discussed in SECY-05-0197 (Ref.9.5.1-2), the fire protection program is an operational program with implementation milestones for various individual elements of the program and applicable codes and standards that apply to the program elements. This section addresses features and elements of the fire protection program that are

- Deluge Sprinkler or Water Spray System – A system employing open sprinklers or spray nozzles (i.e., no fusible link or glass bulb) attached to a dry piping network, with fire detector(s) installed in the same area as the sprinklers/spray nozzles. Operation of the fire detection system opens a deluge valve, which permits water to flow into the sprinkler system network and to be discharged from all the sprinklers or spray nozzles. System operation is terminated manually by shutting the water supply valve.

- Water Mist Fire Suppression System – A water mist system is any fire protection suppression or extinguishing system that relies upon the evaporation of small water droplets to suppress or extinguish a fire. Generally, water mist systems use higher pressure and lower water flow rates than conventional alternatives, but this is not required for classification as a water mist system. Water mist systems as specified for the US-APWR are very similar to traditional sprinklers. Automatic nozzles are used, closed heads are laid out in a manner similar to sprinklers, similar obstruction rules apply, and hydraulic calculations are performed using traditional sprinkler tools. The largest difference is that the water requirements for the mist system are substantially less than the sprinkler system, as much as 90% less. A water mist system is used for MCR staff rooms where excessive water discharge from normal NFPA 13 (Ref.9.5.1-21) or 15 (Ref.9.5.1-22) sprinklers/deluge heads is a concern for safe operation and shutdown of the plant. System operation is terminated manually by shutting the water supply valve.

Automatic Gaseous Suppression Systems

The US-APWR employs several gaseous fire suppression systems in select critical plant areas with heavy fire loading or raised-floor compartments where access for fire fighting may be difficult. For each area where a total flooding gaseous fire suppression system is identified, an environmentally-friendly fire suppression clean agent is used (Novec® 1230 fluid in a 5.6% concentration for cable raised-floor areas, or equal). In conjunction with the gaseous system, an air aspirating, very early warning fire detection system (VESDA® or equal) is used to provide notification of a fire. Such an early notification provides a defense-in-depth fire protection approach for these areas which helps assure adequate fire safety for the areas.

9.5.1.2.6 Fire Detection and Fire Alarm System

Fire detection and alarm systems are provided where required by the FHA, in accordance with the guidance of RG 1.189 (Ref. 9.5.1-12), and NFPA 72 (Ref. 9.5.1-23). Fire detection and alarm systems are generally provided in accordance with the guidance of NFPA 804 (Ref. 9.5.1-14) requirements and guidance as modified by RG 1.189 stipulations (Ref. 9.5.1-12). Fire detectors are to be provided for areas containing safety related equipment and initiate fire alarms.

Fire detectors respond to smoke, flame, heat, or the products of combustion. The installation of fire detectors is in accordance with the guidance of NFPA 72 (Ref. 9.5.1-23) and the manufacturer's recommendations. The selection and installation of fire

The COL Applicant shall provide a milestone for completing ~~is responsible to perform~~ a final FHA and safe-shutdown evaluation based on the final plant cable routing, fire barrier ratings, fire loading, ignition sources, purchased equipment and equipment arrangement. The final FHA and safe-shutdown evaluation shall include a review against the assumptions and requirements stated in the initial FHA and safe-shutdown evaluation provided in the DCD. The final FHA and safe-shutdown evaluation shall also include a detailed post-fire safe-shutdown circuit analysis performed and documented using a methodology similar to that described in NEI 00-01, "Guidance for Post-Fire Safe-Shutdown Circuit Analysis" using as-built data. The final FHA shall be performed and documented as an update to the FSAR COLA application and maintained in the licensing basis for the specific site located plant. (COL Item 9.5(1))

9.5.1.4 Inspection and Testing Requirements

The fire protection systems are inspected and tested prior to initial startup. Preoperational testing is described in Chapter 14, Section 14.2.

The fire pumps are initially tested by the manufacturer in accordance with the guidance of NFPA 20 (Ref. 9.5.1-15) to verify pressure integrity and performance. Periodic testing of fire protection systems during plant operation is primarily governed by applicable NFPA codes and standards in accordance with the guidance of RG 1.189 (Ref. 9.5.1-12).

9.5.1.5 Instrumentation Requirements

Pressure sensors start the fire pumps on decreasing fire main water pressure. Pressure indicators confirm adequate pressures for automatic and manual suppression systems, and selected pressure sensors monitor air pressure in fire suppression piping.

Valve position sensors are used to monitor the position of water supply valves (i.e., serve a supervisory function).

The fire water storage tank, if a fire water storage tank is used, is monitored for level and temperature. The diesel-driven fire pump fuel storage tank, if a diesel driven fire pump is used, is monitored for level.

The fire pumps are operable from the MCR. The run status of the fire pumps are indicated on the display in MCR.

9.5.2 Communication Systems

The communication systems provide for effective intra-plant and plant-to-offsite communications during normal, transient, fire, accidents, off-normal phenomena (e.g., LOOP), and security-related events. The various plant communication systems provide independent, alternate, redundant communication paths to ensure the ability to communicate with station and offsite agencies during all operating conditions.

Some parts of the facility communication systems, related functions and external interfaces are the responsibility of the licensee and are addressed by the COL applicant. These items include the communications aspects of the licensee's security and detection

9.5.2.2.2.1 Standard Telephones

Standard telephones are hardwired to the PABX via outlet points (telephone jacks) and support communications within the plant and offsite. Each telephone consists of a handset and a base. The handset is hardwired to the base or it can be cordless with a short wireless connection to the base.

Standard PABX features include:

- Standard Notification System – Provides a communication link with onsite and offsite personnel.
- Ring down Phone Calling Trees – Provides a method to call and notify multiple parties.
- Standard features found on commercial telephones (speakerphone, message handling, etc.)

9.5.2.2.2.2 Emergency Telephones

Emergency telephones are color-coded, (e.g., red), to distinguish them from normal telephones. These telephones are dedicated and are used for:

- Emergency notification system (ENS - NRC)
- Local/state emergency notification
- Health physics network
- Plant security
- Offsite emergency operations facility (EOF)

~~Communication between the TSC is made using the PABX, station radio system, page and sound powered telephone system (except for the offsite facilities).~~ Communication between the onsite technical center (TSC) and main control room (MCR) may be made using the PABX, station radio system, plant page system and the sound powered telephone system. The sound powered telephone system is an on site system and can not be used to communicate with offsite facilities. The PABX telephone system is also used for notification purposes associated with unauthorized or unconfirmed removal of strategic nuclear material pursuant to the requirement addressed in 10 CFR 73.20(a) (Ref. 9.5.2-23).

9.5.2.2.2.3 PABX Power Source

The PABX is powered from the plant non safety-related load group and consists of independent chargers and batteries for each PABX node. The batteries have the capability to operate the plant telephone system for approximately 8 hours following loss

Plant offsite communications arrangements are site-specific and are described by the COL ~~a~~Applicant. The plant will be provided with multiple offsite communications links such as microwave, hardwired (copper), broadband (cable), fiber optic and direct satellite. These links will include both verbal and data communications. A firewall system is provided to protect the plant broadband systems. The use of these alternate links provides access to the nationwide telephone system. They allow the plant to operate and meet regulatory requirements.

9.5.2.2.5.2 Emergency Communications

Effective emergency onsite and plant-to-offsite communications is provided by the onsite PABX and the offsite emergency response center PABX systems. These systems allow for communications during normal as well as off normal situations including design basis accidents, fire, and LOOP.

The offsite communication system is located in the offsite emergency response center identified in 10 CFR 50.47 (b)(8). It is described by the COL ~~a~~Applicant. The effectiveness of the over all Emergency Response Plan pursuant to 10 CFR 50.47 (b)(8) (Ref. 9.5.2-2) is addressed by the COL ~~a~~Applicant.

The PA/PL, PABX, and plant radio systems are normally used for intra-plant normal and emergency communications with the SPTS providing additional capability and backup.

Radiation and fire alarms have priority over page. When the page system receives alarm inputs from the fire or radiation panels, it automatically provides audible messages and tone annunciation in accordance with specified schedules.

The following radio systems provide both in-plant and plant-to-offsite emergency communications:

- Crisis management radio systems in accordance with the intent of NUREG-0654 (Ref. 9.5.2-24)
- Fire brigade radio system, in accordance with BTP SPLB 9.5-1, position C.5.g(4) (Ref. 9.5.2-25)

The emergency offsite communication system, including the crisis management radio system, is addressed by the COL ~~a~~Applicant. The fire brigade radio system is site-specific, consisting of a base unit, mobile units, and portable units, also is addressed by the COL ~~a~~Applicant.

9.5.2.3 Safety Evaluation

Plant communication systems are not required to mitigate a DBA, however they are important to safety. These systems are needed to support effective normal and off-normal operations as well as to coordinate on-site and off-site responses during abnormal or emergency events. The off-site communications systems within the one-site operations support center provide for emergency response following a design

Each Power Source Fuel Storage Vault (PSFSV) is provided with a vapor and liquid detection system that is equipped with on-site audible and visual warning devices with battery backup.

Each fuel oil storage tank and the transfer pumps are located in vault identified as the PSFSVs and each vault is provided with a manually operated ventilation system for personnel safety to remove any vapors when personnel enter the area. The PSFSV will not have a normally running ventilation system. The ventilation system consists of a supply air opening with a backdraft damper at the ceiling of the vault from the outside, and ducted to the bottom of one side of the vault. This duct will have an in-duct electric heater controlled by a local thermostat in the downstream ductwork. An exhaust fan at the ceiling with a backdraft damper to the outside is ducted to the bottom other side of the vault. This local ventilation system will be turned on locally (or from the MCR) only when personnel are required to enter the area for the performance of surveillances, inspections and maintenance activities.

The in-duct electric heater is provided on the supply air duct so that during the winter, whenever the ventilation system is used the incoming cold outside air is heated and the vault area will be able to be maintained above freezing.

Unit heaters are provided to maintain fuel oil temperature within specification for when the Power Source Fuel Storage Vault temperature may drop below 35ºF. The COL Applicant is to address the need for installing unit heaters in the PSFSV. The concrete pipe chases between the fuel oil tank room and the PS/B is where the fuel oil piping is passing through. Within the concrete pipe chase is a 3-hour fire rated wall that separates the PS/B from the PSFSV. The door and penetrations through this wall are all 3-hour fire rated. One side of the concrete pipe chase is part of the PS/B, which is a normally heated building. The other side of the concrete pipe chase is considered a part of the PSFSV and has the same conditions of the vault area and is one of the locations that would have a unit heater if required as part of the COL Applicant evaluation of extreme cold conditions.

The Fuel Oil Storage Tanks are fabricated of carbon steel material that does not contain Cu or Zn. The exterior and interior surfaces of the fuel oil storage tanks are painted with a primer and finish coat system for corrosion protection of the tank surface. Exterior surfaces of the fuel oil transfer piping are painted for corrosion protection. The interior surfaces of the fuel oil storage tanks are coated with epoxy coating that does not contain Cu or Zn which due to exposure could promote fuel degradation and promote gel formation.

The piping material is ASTM A106, Grade B carbon steel and the valve material is carbon steel (Ref. 9.5.4-9).

Materials used (with proper coating, as necessary) are compatible with fuel oil service.

9.5.4.2.2.2 Gas Turbine Generator Fuel Oil Transfer Skids

Each GTG FOS is serviced by a modularized skid mounted fuel oil transfer assembly, consisting of suction strainers, two fuel oil transfer pumps, a moisture separator, and a fuel filter with the interconnecting piping, valves, and instrumentation. These skids are located in the same compartments as the Fuel Oil Storage Tanks.

The fuel oil transfer pump skids are powered from their respective Class 1E power buses.

The fuel oil storage tanks are sized to provide sufficient capacity for seven days of operation for each GTG. The COL ~~a~~Applicant is to confirm that the operator can arrange for additional fuel to be delivered to the plant site within this period. Each pump is powered from the same train Class 1E bus. Failure of one FOS train would not affect the operability of components in the other train.

The fuel oil temperature is maintained above the cloud point to assure its quality. The fuel oil temperature above the cloud point is achieved by an area electric heater located in the fuel oil storage tank vault, as necessary and by routing of the transfer piping in the ~~underground tunnel~~ concrete pipe chases to the power source building. The fuel oil in the transfer line can be maintained above the cloud point temperature with the area electric heater in service and operation in the recirculation mode (bypassing the day tank) back to the fuel oil storage tank.

The fuel oil storage tank inventory is sampled for quality on a periodic basis for specific gravity, water sediment, viscosity, contamination, algae formation, etc. and if degradation is detected, corrective action is taken, as discussed in subsection 9.5.4.2.3. A flowing sample point is provided on the inlet line to the fuel oil day tank. A tank aggregate sample point is provided from the top of the tank to sample the fuel oil (including sediments and water) in the fuel oil storage tank. These sample points are shown in Figure 9.5.4-1.

The fuel stored in the fuel oil storage and day tanks shall be removed and the accumulated sediment removed and the tanks cleaned every ten year intervals, as a minimum or if fuel oil degradation is detected. The fuel oil from the day tank to be cleaned will be drained to the fuel oil storage tanks. The fuel oil from the storage tank to be cleaned will be pumped to an empty tanker and the accumulated sediments drained or removed manually, as necessary and collected in proper containers.

The PSFSV ventilation system is classified as equipment class 5 (Non-Safety) and seismic category II. This equipment is in a seismic category I structure with equipment classified as safety-related.

The ventilation openings at the ceiling will have a seismic missile enclosure to protect thesafety-related fuel oil tank. The ventilation backdraft dampers and exhaust fans for ease of access for maintenance are to be located within these missile enclosures. The backdraft damper is designed to withstand the effects of a tornado.

9.5.4.4 Inspection and Testing Requirements

The FOS is tested prior to initial startup. Preoperational testing is described in Section 14.2. System performance is verified during periodic GTG testing. Periodic inspection of the fuel oil storage tank vents is performed to assure there are no obstructions.

Inservice inspection of piping is performed in accordance with the requirements of ASME Section XI, as discussed in Section 6.6 (Ref. 9.5.4-11).

Technical Specification surveillance testing and inspection of the FOS is performed to assure operational readiness, without loss of system function as described in Chapter 16.

exhaust system are designed so that failure cannot impair the functioning of safety-related equipment.

C. A single failure is assessed as a failure of the GTG with which the component is associated. In such a circumstance, safe shutdown is attained and maintained by the redundant GTG installation.

D. Cooling air for the GTG and room ventilation is drawn through a separate duct. The cooling/ventilation air is exhausted through a separate return duct system. A variable damper is installed in the air exhaust duct and their position will be aligned and set at the installation to relieve air pressure in the room.

9.5.8.4 Inspection and Testing Requirements

The combustion air intake and exhaust system is tested prior to initial startup. Preoperational testing is described in Section 14.2. System performance during normal operation is verified. The ventilation and cooling functions of the GTG combustion air intake and exhaust system are also tested as part of Class 1E GTG testing described in Subsection 14.2.12.1.44.

Inservice inspection of piping is performed in accordance with the requirements of ASME Section XI, as discussed in Section 6.6 (Ref. 9.5.4-11).

9.5.8.5 Instrumentation Requirements

The GTG combustion air intake and exhaust system is provided with instrumentation consisting of a combustion air pressure indicator and exhaust gas temperature indicators. The GTG room is provided with thermometers to monitor room and air exhaust temperature, ventilation / cooling air flow meter.

Thermocouples are used to sense turbine exhaust gas temperature and the turbine exhaust stack temperature. A digital temperature indicator with manual selector switch is located at the GTG control cabinet for selecting turbine exhaust stack temperature. At 100% rated load, the exhaust stack temperature is approximately 1,100 ºF + 50 ºF.

9.5.9 Combined License Information

COL 9.5(1) *The COL ~~a~~Applicant establishes a fire protection program, including organization, training and qualification of personnel, administrative controls of combustibles and ignition sources, firefighting procedures, and quality assurance.*

COL 9.5(2) *The COL Applicant addresses the design and fire protection aspects of the facilities, buildings and equipments, such as cooling towers and a fire protection water supply system, which are site specific and/or are not a standard feature of the US-APWR.*

COL 9.5(3) The COL Applicant describes the provide~~s~~d apparatus for plant personnel and fire brigades such as portable fire extinguishers and self contained breathing apparatus.

COL 9.5(4) The COL Applicant addresses all communication system interfaces external to the plant (offsite locations). These include interfaces to utility private networks, commercial carriers and the federal telephone system. The configuration of these connections will include consideration of the concerns raised in IE Bulletin 80-15.

COL 9.5(5) The COL Applicant addresses the emergency offsite communications including the crisis management radio system.

COL 9.5(6) The COL Applicant addresses connections to the Technical Support Center from where communications networks are provided to transmit information pursuant to the requirements delineated in 10 CFR 50 Appendix E, Part IV.E.9.

COL 9.5(7) The COL Applicant addresses a continuously manned alarm station required by 10 CFR 73.46(e)(5) and the communications requirements delineated in 10 CFR 73.45(g)(4)(i) and (ii). The COL Applicant addresses notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material in accordance with 10 CFR 73.45(e)(2)(iii).

COL 9.5(8) The COL Applicant addresses offsite communications for the onsite operations support center.

COL 9.5(9) The COL Applicant addresses the emergency communication system requirements delineate in 10 CFR 73.55(f) such that a single act cannot remove onsite capability of calling for assistance and also as redundant system during onsite emergency crisis.

COL 9.5(10) Deleted

COL 9.5(11) The COL Applicant is to specify that adequate and acceptable sources of fuel oil are available, including the means of transporting and recharging the fuel storage tank, following a design basis accident.

COL 9.5(12) The COL Applicant is to address the need for installing unit heaters in the Power Source Fuel Storage Vault during the winter for site locations where extreme cold temperature conditions exist.

9.5.10 References

9.5.1-1 "Fire protection," Energy. Title 10 Code of Federal Regulations Part 50.48, U.S. Nuclear Regulatory Commission, Washington, DC.

Table 9.5.1-1 US-APWR Fire Protection Program Conformance with RG 1.189 (Sheet 40 of 46)

Regulatory Position	Position Number	Conformance	Remarks
Pump houses and rooms housing redundant pump trains important to safety should be separated from each other and from other areas of the plant by fire barriers having at least 3-hour ratings. These rooms should be protected by automatic fire detection and suppression unless a fire hazards analysis can demonstrate that a fire will not endanger other equipment required for safe plant shutdown. Fire detection should alarm and annunciate in the MCR and alarm locally. Hose stations and portable extinguishers should be readily accessible.	6.1.9	Conform	Rooms have fire detection installed. Automatic suppression is not provided unless there is significant lube oil associated with the unit based upon the FHA (See Appendix 9A).
Other areas within the plant may contain hazards or equipment that warrant special consideration relative to fire protection, including areas containing significant quantities of radioactive materials, yard areas containing water supplies or systems important to safety, and the plant cooling tower.	6.2	Informational Statement	
New Fuel Areas. Portable hand extinguishers should be located near this area. In addition, hose stations should be located outside but within hose reach of this area. Automatic fire detection should alarm and annunciate in the MCR and alarm locally. Combustibles should be limited to a minimum in the new fuel area. The storage area should be provided with a drainage system to preclude accumulation of water.	6.2.1	Conform, COL for combustible controls.	The COL ~~a~~Applicant establish combustible control procedures. COL item 9.5(1)
Spent Fuel Areas. Local hose stations and portable extinguishers should provide protection for the spent fuel pool. Automatic fire detection should alarm and annunciate in the MCR and to alarm locally.	6.2.2	Conform	

Chapter 10

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10-viii	Figure List	Correction (editorial corrections) Correct the title of Figure 10.2-1 from "Summer" to "Rated".
10.1-7	Figure 10.1-2	Correction (editorial corrections) Correct the title of Figure 10.2-1 from "Summer" to "Rated".
10.2-16	Subsection 10.2.3	Other Added the last sentence as shown below: For the verification that actual rotor material properties satisfy the material properties assumed and used in the turbine missile calculations, mechanical properties including fracture toughness are to be verified by the tests to conform to the applicable material specifications of turbine missile calculations. (Reference 10.2-9).
10.2-16	Subsection 10.2.3.1	Added “are” and “A minimum of three Charpy test specimens are tested using the impact test criteria that satisfy ASTM A470 Grade C (Class 6).” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
10.3-2	10.3.1.1	Replaced “including” with “excluding” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
10.3-4	Subsection 10.3.1.2	Correction (editorial corrections) Corrected the 4th bullet of the 1st paragraph as shown below: The MSS together with the turbine bypass system is capable of accepting 100% load rejection without reactor trip and without lifting MSSVs **and MSRVs**.
10.3-4	Subsection 10.3.1.2	Correction (editorial correction) Corrected the turbine bypass capacity from “67 %” to “67.5 %” in the 5th bullet of the 1st paragraph.
10.3-5	Subsection 10.3.2.2.1	Correction (editorial corrections) Corrected the 4th sentence of in the 1st paragraph as shown below: Velocities are ~~less than~~ approximately 150 ft/sec.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.3-7	Subsection 10.3.2.3.2	Correction (editorial corrections) Corrected the 2nd paragraph as shown below: The total **<u>required</u>** capacity of these valves is 105% of the main steam flow rate at rated power conditions.
10.3-9	Subsection 10.3.2.3.4	Correction (editorial corrections) Corrected the 2nd paragraph as shown below: If the MSLB occurs upstream of MSIVs, ~~even if a single failure of this valve is assumed,~~ the broken side SG is isolated by the MSIVs on the main steam piping of the intact SGs or the **<u>MSIV/</u>**MSCV of the broken line. In case of a line break downstream of the MSIV, ~~even if a single failure of this valve is assumed,~~ MSIVs on the main steam piping of the **<u>both</u>** intact SGs **<u>and faulted SG</u>** would prevent the steam blowdown ~~through more than one SG~~. **<u>See Table 10.3.3-1 which shows failure of either train solenoid valve for MSIV actuation does not impair isolation function of MSIV.</u>**
10.3-10	Subsection 10.3.2.4.2	Correction (editorial corrections) Corrected the 2nd paragraph as shown below: In the event of a design-basis accident, such as a main steam line break, the MSIVs with associated MSBIVs are automatically closed. In case the line break is downstream of the MSIV, ~~even if a single failure of this valve is assumed,~~ the MSIVs on the main steam piping of the **<u>both</u>** intact SGs **<u>and faulted SG</u>** would prevent the steam blowdown through more than one SG. **<u>If the MSLB occurs in the upstream of MSIVs, the broken side SG is isolated by the MSIVs on the main steam piping of the intact SGs or the MSIV/MSCV of the broken line.</u>**
10.3-24	Table 10.3.2-2	Detailed engineering progress Changed the relieving capacity per main steam line to “5,304,000 (lb/hr)”. Changed the total relieving capacity to “21,216,000 (lb/hr)”.
10.3-26	Table 10.3.2-2 Main steam bypass isolation valves	Correction (editorial corrections) Corrected the valve size of main seam bypass isolation valves to “4 (in)”.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-1	Subsection 10.4.1.1.2	Correction (editorial correction) Corrected the turbine bypass capacity from “67 %” to “67.5 %” in the 2nd bullet of the first paragraph.
10.4-2	Subsection 10.4.1.2.1	Correction (editorial correction) Corrected the turbine bypass capacity from “67 %” to “67.5 %” in the 5th paragraph.
10.4-5	Table 10.4.1-1	Correction (editorial corrections) Correct the tube thickness from "22BWG" to "23BWG"
10.4-10	Figure 10.4.2-1	Engineering progress and suppliers information Remove seal water valve symbol from fifteen valves between condensers and strainers.
10.4-10	Figure 10.4.2-1	Correction (editorial corrections) Correct the direction of a reducer in the condenser c left side outlet line.
10.4-14	Figure 10.4.3-1	Correction (editorial corrections) Correct system code from "TDS" to “TVD” in the grand steam condenser exhaust fans outlet line.
10.4-15	Subsection 10.4.4.1.1	Correction (editorial correction) Corrected the turbine bypass capacity from “67 %” to “67.5 %” in the 1st bullet of the 1st paragraph.
10.4-17	Subsection 10.4.4.4	Correction (editorial correction) Corrected the description of the 3rd sentence of the 4th paragraph regarding turbine bypass capacity which is corrected to 67.5% as shown below: Three TBVs with 13.4**<u>5</u>** % of rated main steam flow of 20,200,000 lb/h at ~~the~~**<u>a</u>** valve inlet pressure **<u>of 792 psia</u>**~~774 psig~~ perform adequate decay heat removal to keep the cooldown rate of reactor coolant system at 50 deg.F/h during normal plant shutdown and thereby reduce the demands on systems important to safety in meeting GDC 34.
10.4-18	Table 10.4.4-1	Correction (editorial correction) Corrected the turbine bypass valve capacity to 909,000 lb/hr. Corrected the total capacity of the turbine bypass valves to 13,635,000 lb/h.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-20	Subsection 10.4.5.2.1	Correction (editorial corrections) Correct the numbers of non-essential service water pumps.
10.4-27	Table 10.4.5-1	Other Material information is added for the CWS above ground piping.
10.4-33-34	Table 10.4.6-1	Correction (editorial corrections) Delete the design pressure and temperature condition unconformity.
10.4-39	Subsection 10.4.7.1.2	Correction (editorial correction) Corrected the last sentence of the 18th paragraph as shown below: During ~~normal or~~ upset **<u>or abnormal</u>** conditions, the function of these check valves is to prevent reverse flow from the SGs whenever the FWS is not in operation. Corrected the 3rd sentence of the last paragraph as shown below: The failure of one solenoid valve **<u>does</u>** not impair the isolation function of MFIV.
10.4-43	Subsection 10.4.7.2.2	Correction (editorial correction) Corrected the 14th paragraph as shown below: MFRV is designed to close within 5 seconds after receiving signals, such as an ECCS actuation signal, high-high SG water level signal, ~~and~~ P-4 & low Tavg signal, **<u>and high SG water level signal</u>**. Details of the three element control system are provided in Chapter 7. Corrected the 17th paragraph as shown below: The MFBRV is designed to close within 5 seconds after receiving signals, such as a ECCS actuation signal, ~~or,~~ a high-high SG water level **<u>signal and high SG water level signal</u>**.
10.4-55	Table 10.4.7-2	Correction (editorial corrections) Correct the motor types from "Synchronous" to "Induction" for the pumps.
10.4-61	Figure 10.4.7-4	Engineering progress and suppliers information Pressure indicators on feedwater booster pump inlet lines are moved from strainer upstream to downstream

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-61	Figure 10.4.7-4	Engineering progress and suppliers information Change valves from manual type to motor type on feedwater heater 6A/B inlets and feedwater heater 7A/B outlets.
10.4.64	10.4.8.2.1	Deleted “During plant startup w” and added “W” in the 7th paragraph
10.4-65	Subsection 10.4.8.2.1 16th paragraph	Engineering progress Change of the sentence about SG blow-down water quality monitoring system because the design shall be changed based on EPRI guideline
10.4-66	Subsection 10.4.8.2.1 17th paragraph 5th and 6th bullet	Engineering progress Delete the signals of high water level and high pressure in the blowdown flash tank
10.4-67	Subsection 10.4.8.2.2.4	Engineering progress Add the sentence about draining method for remaining water
10.4-71	10.4.8.5	“control” was replaced with “ondicate” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
10.4-73	Table 10.4.8-1 (Sheet 1 of 3)	Engineering progress The new DCD parameters show design parameters (i.e., process parameters) while DCD Rev.2 shows the parameters used for safety evaluation.
10.4-74	Table 10.4.8-1 (Sheet 2of 3)	Engineering progress The new DCD parameters show design parameters (i.e., process parameters) while DCD Rev.2 shows the parameters used for safety evaluation.
10.4-79	Subsection 10.4.9	Correction (editorial corrections) Corrected the 2nd sentence of the 2nd paragraph as shown below: The EFWS is a**<u>n</u>** ASME Code, Section III (Reference 10.4-8), Class**<u>es 2 and</u>** 3, Seismic Category I, redundant system with Class 1E electric components as indicated in Table 3.2.2.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-80	Subsection 10.4.9.1	Correction (editorial corrections) Corrected the 12th bullet of the 1st paragraph as shown below: • The EFWS design is provided with the capability to automatically terminate EFW flow to a depressurized (faulty) SG and to automatically provide EFW to the intact SGs. **<u>The EFWS design is also capable of automatically terminating EFW flow to prevent overfilling of the SGs.</u>**
10.4-81	Subsection 10.4.9.1	Correction (editorial corrections) Corrected the 20th bullet of the 1st paragraph as shown below: • The EFW pump main steam line steam isolation valves (containment isolation **<u>v</u>**~~c~~alve) in the steam supply lines and the steam piping upstream of the containment isolation valves are ASME Code, Section III (Reference 10.4-8), Class 2. The steam supply lines to the EFW pump turbine from ~~steam lines~~**<u>the</u>** downstream of the containment isolation valves are designed and constructed in accordance with ASME Code, Section III (Reference 10.4-8), Class 3 requirements.
10.4-82	Subsection 10.4.9.1	Added below 23rd bullet. • The recommendations of RGs 1.36 and 1.37 are applied during fabrication of the EFWS and preheat guidelines in ASME Code Section III, Appendix D, Article D-1000 for carbon steel are applied to the EFS component.
10.4-82	Subsection 10.4.9.2	Correction (editorial corrections) Corrected the 2nd sentence of the 8th paragraph as shown below: As a result, **<u>almost</u>** none of the EFW pump flow is lost by spilling out of the break.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-83	Subsection 10.4.9.2.1	Correction (editorial corrections) Corrected the 3rd paragraph as shown below: The EFW pump is designed to develop adequate head to supply the design flow of at least 400 gpm to each SG, when the SG pressure is equivalent to the set pressure of the first stage of the main steam safety valve (safety valve with lowest set pressure) plus 3% of accumulation **and the pump discharge tie line is closed**.
10.4-84	Subsection 10.4.9.2.1	Correction (editorial corrections) Corrected the 9th sentence to 12th sentence of the 11th paragraph as shown below: The inside dimensions of each pit is approximately 28 feet long, approximately ~~42~~**43** feet wide and approximately 35 feet **deep**~~depth~~. With the minimum pit level ~~at~~**of 92.5%** ~~approximately 26 feet~~ during normal plantcondition, the volume of water in each pit available for the EFW is 186,200 gallon. With two pits, each with a capacity of **241,000**~~204,850~~ gallons **at the water levels from 0 to 100%**, it is sufficient to perform hot standby and plant cooldown until the RHRS starts to perform heat removal. And also each pit has adequate capacity ~~for~~**from** the pit low level alarm setpoint to allow at least 20 minutes for operator action in accordance with the additional short-term recommendation “Primary EFW Water Source Low Level Alarm,” of generic recommendations of NUREG-0611 and NUREG-0635.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-85	Subsection 10.4.9.2.1	Correction (editorial corrections) Corrected the 2nd paragraph as shown below: The makeup line routed from the demineralized water storage tank to the EFW pit is used for initial water fill of the EFW pits and to provide makeup water to maintain the water level in the EFW pits during normal plant operation.The demineralized water storage tank provides a backup source for EFWS. Due to a sufficient volume of water in the EFW pits for safe shutdown ~~of~~ **<u>by</u>** keeping the plant at hot standby for 8 hours and performing plant cooldown to RHR entry condition for 6 hours after accident or transient, this backup supply is not required to be safety-related.The manual valves from the demineralized water storage tank to the EFW pumps are normally closed. If the water level of both EFW pits reaches low-low water level after an accident or transient without stabilizing at MODE 4 condition, the manual isolation valve will be opened by an operator. Before opening the isolation valve, the operator **<u>will</u>** verify that the storage tank has adequate water level to keep sufficient NPSH of the EFW pumps. Corrected the 14th paragraph as shown below: Because the EFW pits have the water supplied directly from **<u>demineralized water</u>** ~~condensate~~ storage tank without deaerating and the inventory water of the pit has direct contact with atmosphere, the dissolved oxygen level of the pit inventory is not zero, however, because the design temperature of the EFW pit is ~~105~~**<u>150</u>** Deg F, which is determined to exceed assumed maximum operating temperature of the EFWS, ~~the~~ stress corrosion cracking would not occur in such low temperature condition even if the level of dissolved oxygen is high, therefore, the EFW pits have adequate integrity.
10.4-85	Subsection 10.4.9.2.1	Correction (editorial corrections) Corrected the 15th paragraph as shown below: Sampling of the EFW pits is performed monthly, and turbidity is ensured to be not over 1 ppm. Any deviation is corrected by utilizing ~~bleed~~**<u>feed</u>** and ~~feed~~**<u>bleed</u>** method. Demineralized water from the demineralized water storage tank (make-up water source) is used for feeding the water inventory. Complete inspections with the pits drained will be performed periodically per the ISI program.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-85	Subsection 10.4.9.2.1	Correction (editorial corrections) Added the 2nd sentence of the 17th paragraph as shown below: **<u>The motor-operated valves are also closed on receipt of such signal as high SG water level or low main steam line pressure.</u>**
10.4-86	Subsection 10.4.9.2.1	Correction (editorial corrections) Corrected the 4th sentence of the 18th paragraph as shown below: They are closed if required to terminate a leak or break or ~~if~~**<u>when</u>** the EFW pump actuation valve requires maintenance.
10.4-86	Subsection 10.4.9..2.1	Detailed engineering progress Revised the 19th paragraph to reflect the addition of emergency feedwater pump actuation valves. IST of the T/D-EFW pump actuation valve (MOV-103A(D)) can be performed during A(D) train OLM to preclude unnecessary actuation of T/D-EFW pump. However, full stroke/quarterly operability IST exercise requirement for MOV-103A(D) requires quarterly OLM for A(D) train. Further, for the IST during non-OLM of the pump, closing MOV-101A© and B(D) to prevent the pump actuation will reduce the number of available EFW pumps leading to deviation from EFWS LCO (T-spec 3.7.8). Moving the 103A and the 103D to downstream of the 101A and 101D but upstream of connection point form each A and B main steam piping respectively, and adding another actuation valve downstream of the MOV-101B and the 101C allow performance of the actuation valves IST without affecting OLM.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-88	Subsection 10.4.9.2.2	Correction (editorial corrections) Corrected the last sentence of (e) Main Steam Line Break (MSLB) as shown below: The EFW **supply** function is not needed during the mitigation of the MSLB accident~~, but is needed only for cooldown up to the RHR system initiation~~. Corrected (f) of the 15th sentence as shown below: A SBO results in the loss of normal offsite and emergency onsite ac power sources.The M/D-EFW pumps are inoperable because there is no ac power. Both T/D EFW pumps are available because of the dc power supplied by class 1E batteries with two hours capacities. EFW flow control is also available because the EFW flow control valves are powered by dc power which is available from class 1E batteries. In addition, at least within one hour after the SBO occurrence, one unit of the AAC-GTG is started, and by the operation of one unit of emergency feedwater pump (turbine-driven) area air handling units, the integrity of one unit of T/D EFW pump is ensured. The AAC-GTGs minimize the potential for common cause failures with the Class 1E GTG as discussed in Section 8.4.1.3. From the above, because the AAC GTGs are available during SBO event, in accordance with the generic recommendations of NUREG-0611 and NUREG-0635 Generic Short Term Recommendation No. 5 (GS-5), the EFWS is capable of providing required EFW flow for at least two hours from one T/D-EFW pump. After starting the operation of the AAC-GTG, charging to the Class 1E batteries ~~are~~**is** resumed, therefore, the turbine-driven EFW pump is able to continue to operate after two hours of the SBO and is independent of any ac power source.
10.4-89	Subsection 10.4.9.2.2	Correction (editorial corrections) Corrected the 3rd sentence of (h) Steam Generator Tube Rupture (SGTR) as shown below: Upon detection of a water level increase in the faulted SG, the EFW isolation valve **on piping** to the **faulted**~~all~~ SG is automatically closed.
10.4-91	10.4.9.2.4	RAI No. 423-2710 REVISON 1, Question No. 19-375 Adding the description regarding non safety-related EFW pit water level.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-92	10.4.9.3	Detailed engineering progress Deleted available NPSH and required NPSH. These information are detail design scope.
10.4-95	Table 10.4.9-1	Detailed engineering progress Deleted the following information which are detail design scope: • NPSH available maximum operating flow (ft) • Material specification • Motor specification
10.4-95	Table 10.4.9-1	Correction (editorial corrections) Corrected the total dynamic head of the motor-driven emergency feedwater pump.
10.4-96	Table 10.4.9-1	Detailed engineering progress Deleted the following information which are detail design scope: • NPSH available maximum operating flow (ft) • Material specification
10.4-96	Table 10.4.9-1	Correction (editorial corrections) Corrected the total dynamic head of the turbine-driven emergency feedwater pump.
10.4-97	Table 10.4.9-1	Correction (editorial corrections) Corrected the following information: • EFW pit inside dimensions • Design temperature of EFW control valves and EFW isolation valves.
10.4-97	Table 10.4.9-1	“42” was changed to “43” “105” was changed to “150” “568” was changed to “150” “2” was changed to “3” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-98	Table 10.4.9.1	Correction (editorial corrections) Adding missing tables for turbine-driven EFW pump main steam-line steam isolation valves and turbine-driven EFW pump actuation valves which are described to be referred to in Subsection 10.4.9.2.1.
10.4-99	Table 10.4.9-3	Correction (editorial corrections) Adding a note as shown below: **<u>(Note) Initial flow rates to SGs are shown. The flow rates will be decreased to prevent SG overfilling and to keep SG water level.</u>**
10.4-101	Table 10.4.9-4	Detailed engineering progress Adding EFW-MOV-103B and C to improve the convenience for the EFW actuation valves IST.
10.4-102	Table 10.4.9-4	Detailed engineering progress Adding EFW-MOV-103B and C to improve the convenience for the EFW actuation valves IST.
10.4-103	Table 10.4.9-5	RAI No. 423-2710 REVISON 1, Question No. 19-375 Adding the description regarding non safety-related EFW pit water level.
10.4-104 to 105	Table 10.4.9-6	Detailed engineering progress Adding EFW-MOV-103B and C to improve the convenience for the EFW actuation valves IST.
10.4-123	Subsection 10.4.11.2.1	Correction (editorial corrections) Corrected the 1st paragraph as shown below: The ~~conceptual flow~~ **<u>ASSS piping and instrumentation</u>** diagram is shown in Figure 10.4.11-1.
10.4-127	Figure 10.4-11	Correction (editorial correction) Corrected the title of figure as shown below: Figure 10.4**<u>.</u>**-11**<u>-1</u>** Auxiliary Steam Supply System Piping and Instrumentation Diagram

Figures

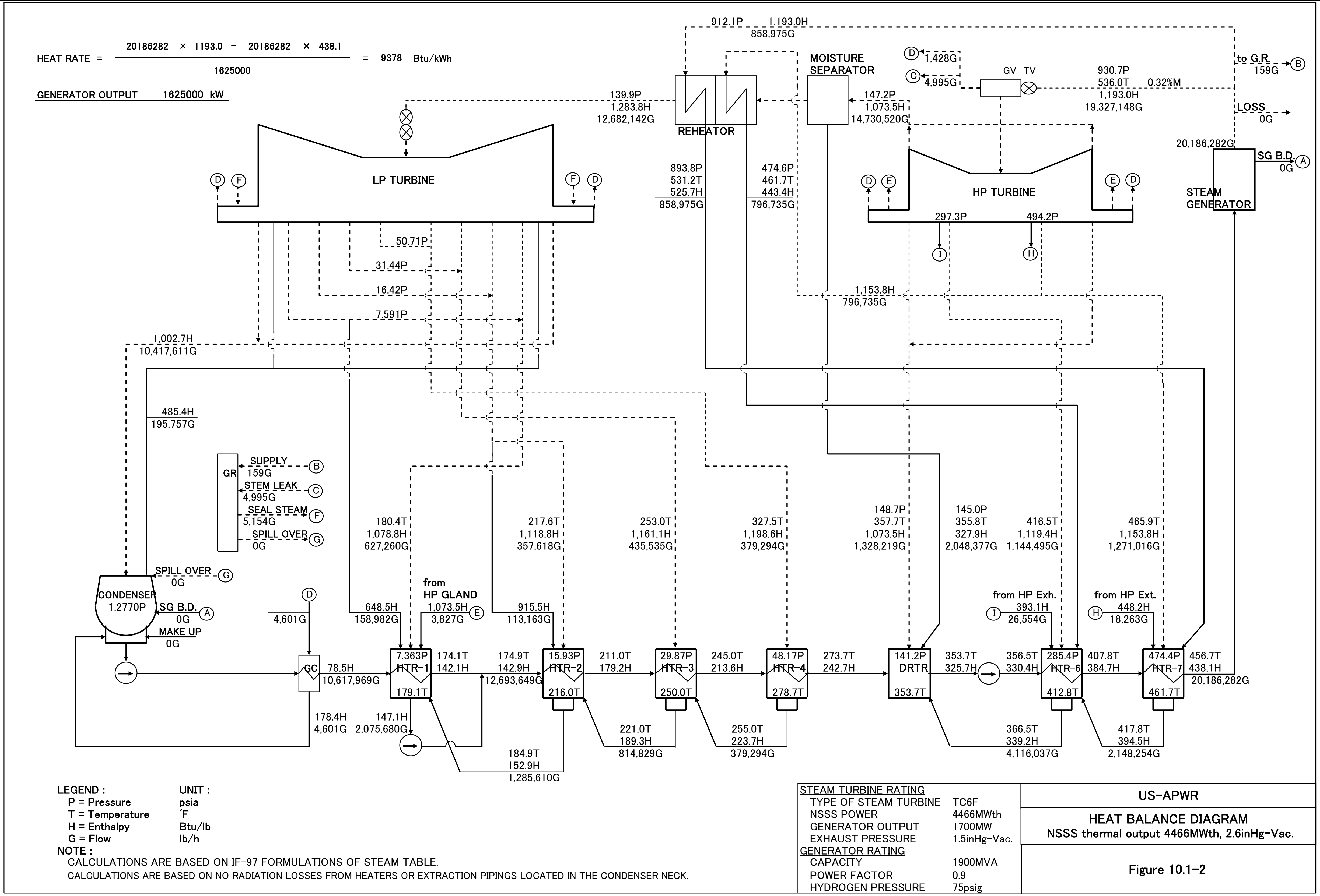


Figure 10.1-2 Heat Balance Diagram ~~Summer~~ Rated Condition (Cond. vac. : 2.6inHgA)

10.2.3 Turbine Rotor Integrity

Turbine rotor integrity is provided by the integrated combination of material selection, rotor design, fracture toughness requirements, tests, and inspections. This combination results in a very low probability of a condition that could result in a rotor failure. For the verification that actual rotor material properties satisfy the material properties assumed and used in the turbine missile calculations, mechanical properties including fracture toughness are to be verified by the tests to conform to the applicable material specifications of turbine missile calculations. (Reference 10.2-9)

10.2.3.1 Materials Selection

Fully integral turbine rotors are made from ladle refined, vacuum deoxidized Ni-Cr-Mo-V alloy steel by processes that maximize the cleanliness and toughness of the steel. The lowest practical concentrations of residual elements are obtained through the melting process. The turbine rotor material complies with the chemical property limits of ASTM A470 (Reference 10.2-5). The specification for the rotor steel has lower limitations than indicated in the ASTM standard (Reference 10.2-5) for phosphorous, sulphur, aluminum and antimony. This material has the lowest fracture appearance transit temperatures (FATT) and the highest Charpy V-notch energies obtainable on a consistent basis from water-quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Charpy tests and tensile tests are in accordance with ASTM, A370 (Reference 10.2-6). A minimum of three Charpy test specimens are tested using the impact test criteria that satisfy ASTM A470 Grade C (Class 6).

The production of steel for the turbine rotors starts with the use of high-quality, low residual element scrap. An oxidizing electric furnace is used to melt and dephosphorize the steel. Ladle furnace refining is then used to remove oxygen, sulphur, and hydrogen from the rotor steel. The steel is then further degassed using a process whereby steel is poured into a mold under vacuum to produce an ingot with the desired material properties. This process minimizes the degree of chemical segregation since silicon is not used to deoxidize the steel.

10.2.3.2 Fracture Toughness

Suitable material toughness is obtained through the use of materials described in Subsection 10.2.3.1 to produce a balance of material strength and toughness to provide safety while simultaneously providing high reliability, availability, and efficiency during operation. The restrictions on phosphorous, sulphur, aluminum, antimony, tin, argon, and copper in the specification for the rotor steel provides for the appropriate balance of material strength and toughness. The impact energy and transition temperature requirements are more rigorous than those given in ASTM A470 Class 6 or 7 and their equivalents.

Stress calculations include components due to centrifugal loads and thermal gradients where applicable. Fracture toughness will be at least 200ksi·in$^{1/2}$ (220MPa·m$^{1/2}$). For the purpose of conservative evaluation, fracture analysis is to be done using a fracture toughness with margin against minimum expected values on the rotors. The material

Conformance to GDC 34 (Reference 10.3-1) assures redundant cooling capacity and the pressure relief capability of the MSS in conjunction with emergency feedwater system so that the components retain their safety functions in the event of single component failures.

In conformance with Regulatory Guide 1.155 (Reference 10.3-2), “Station Blackout”, and in compliance with 10 CFR 50.63 (Reference 10.3-3), the US-APWR is provided with an AAC (alternate ac) power source to cope with an SBO event. Refer to Section 8.4 for further details.

Conformance to Regulatory Guide 1.115 (Reference 10.3-4), “Protection Against Low-Trajectory Turbine Missiles”; Regulatory Guide 1.117, “Tornado Design Classification” for protection against tornadoes; and Regulatory Guide 1.29, “Seismic Design Classification” (Reference 10.3-5) that reflects US-APWR equipment class are demonstrated and discussed in Sections 3.5, 3.3, and 3.2, respectively.

Codes and standards used in the design of the MSS, quality group and seismic classification are identified in Section 3.2. The following MSS components are classified as Equipment Class 2, and are safety-related and are designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III (Reference 10.3-6), Class 2, seismic category I:

- All piping and valves from the SGs up to and including MSIV and MSBIV.
- Branch lines from the above described main steam piping up to and including, the first valve, which includes MSSV.
- Inlet piping from the main steam line up to and including MSRVs and MSDVs.
- Branch lines from the main steam piping to the emergency feedwater pump turbines up to and ~~including~~excluding the first motor-operated valve.
- Main steam drain piping upstream of MSIV up to and including main steam drain line isolation valves (MSDIVs).
- Nitrogen supply line located on the main steam piping upstream of MSIV, up to and including the first isolation valve.

The following MSS components are classified as Equipment Class 3, and are safety-related and are designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III (Reference 10.3-6), Class 3, seismic category I:

- MSS piping downstream of MSIV and MSBIV up to and including the first restraint located in the main steam/feedwater piping area.
- MSSV, MSRV and MSDV discharge piping located in the main steam/feedwater piping area.
- Downstream piping of MSDIVs located in the main steam/feedwater piping area.

10.3.1.2 Non Safety Power Generation Design Bases

The following is a list of the non safety power generation design bases:

- The MSS is designed to deliver steam from the SGs to the steam turbine generator for the range of flow rates, temperatures and pressures from warming of the main steam piping to rated power conditions.

- Each main steam line is sized to provide balanced steam pressures to the main turbine stop valves. The main steam equalization piping located midway between these lines is designed to equalize the pressure from individual main steam lines.

- The MSS is capable of accepting a ± 10% step load change and a ± 5%/min ramp load change without discharging steam to the condenser or the atmosphere. For large load change step reductions, steam is bypassed directly to the condenser via the turbine bypass system.

- The MSS together with the turbine bypass system is capable of accepting 100% load rejection without reactor trip and without lifting MSSVs and MSRVs.

- The MSS provides the capacity to dump 67.5% of rated power steam flow to the condenser resulting from 100% load reduction.

- The MSS provides the means of dissipating residual and sensible heat generated from the NSSS during hot standby and cooldown even when the main condenser is not available. MSDVs or MSRVs are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.

- The MSS provides the ability to dry and reheat the exhaust steam from the high-pressure turbine and delivers steam to the low-pressure turbine.

- The MSS design prevents water induction into the turbine during transient conditions. The MSS also provides turbine over speed protection during transient conditions by limiting stored energy in feedwater heaters.

- The MSS collects the drainage condensed in the main steam and reheat piping, and transports it to the condenser.

10.3.2 Description

10.3.2.1 General Description

The MSS is primarily a steam transport system consisting of piping and valves and associated instrumentation. MSS piping and components are located within the containment, in the main steam/feedwater piping area in the reactor building and the turbine building. The MSS piping and instrumentation diagrams are shown in Figures 10.3-1, 10.3-2, 10.3-3 and 10.3-4. Table 10.3.2-1 provides MSS performance data. The system includes the following major components:

- Main steam piping from the SG outlet steam nozzles to the main turbine stop valves
- MSIV and MSBIV in each main steam line
- Main steam check valve (MSCV) in each main steam line
- MSSVs, MSRV and MSDV in each main steam line
- Main steam relief valve block valve (MSRVBV) in each main steam line
- Main steam branch line from each main steam line to emergency feedwater pump turbine
- TBVs (see Subsection 10.4.4)

10.3.2.2 Main Steam System – Detailed Description

10.3.2.2.1 Main Steam Delivery

The MSS transports and distributes steam from the SG system to the main turbine system (MTS) during power generation and directly to the main condenser when the MTS is not available. The piping is designed such that the pressure drop from the SG to the turbine main steam stop valve does not exceed 41.3 psi at rated power steam flow conditions. The low-pressure drop assures the steam moisture content does not exceed 0.5%. Piping is sized for rated power steam flow conditions. Velocities are ~~less than~~ approximately 150 ft/sec. These are four, 32 inch diameter main steam lines, one from each SG supplies steam to the turbine generator. The 32 inch diameter main steam lines from the SGs are connected to the 42 inch equalization piping located near, but below, the high-pressure turbine. The portion of the steam from the equalization piping flows to gland steam seals, moisture separator reheater, deaerator heating with the high-pressure turbine receiving the balance of the flow via four individual lines and four main turbine stop and control valves. The main turbine stop valves and main turbine control valves are part of the MTS and are discussed in Section 10.2. Each of the main steam lines is anchored in the main steam/feedwater piping area adjacent to the turbine building.

The sizes and layout of the main steam piping from individual SGs hydraulically balances the pressure drop such that the differential pressure between any two SGs does not exceed 10 psi.

Main steam branching from the equalization piping supplies reheating steam to the MS/R 2nd stage tube bundle. Control valves in the reheating steam supply lines control the steam flow to the tube bundles during plant startup and shutdown. Power operated isolation valves and bypass valve are also located in the MS/R reheating steam supply lines.

Connections allowing sampling are provided in appropriate locations in the secondary side piping. The sampling system is described in Subsection 9.3.2

exclusion zone. Section 3.6 addresses the applicability of leak before break and break exclusion zone to the main steam line. This piping is designed to Seismic Category I requirements.

Each SG outlet nozzle is equipped with a flow restrictor to limit the flow in the event of a steam line break. This flow restrictor is a multi-flow nozzle-type with a throat diameter of equivalent to16 inches.

Main steam piping is designed to minimize the effects of erosion/corrosion. Pipe material, pipe wall thickness, fluid velocity, fluid chemistry and piping arrangement affect erosion/corrosion damage.

The main steam piping to the turbine is sized to limit velocities to minimize potential erosion and routed to minimize bends/elbows. Selected pipe wall thickness includes corrosion allowance, accounting for the design life of the plant and pipe wall thickness inspections are performed to monitor wall erosion.

Design parameters for the main steam piping are provided in Table 10.3.2-1 and 10.3.2-3.

10.3.2.3.2 Main Steam Safety Valves

MSSVs with sufficient rated capacity are provided to prevent the steam pressure from exceeding 110 percent of the MSS design pressure:

The total required capacity of these valves is 105% of the main steam flow rate at rated power conditions.

MSSV rated capacity is tabulated in Table 10.3.2-2.

Six MSSVs are provided per main steam line. Table 10.3.2-2 provides performance data and set pressure for the MSSVs.

The MSSVs are located in the safety-related portion of the main steam piping upstream of the MSIVs and outside the containment in the main steam/feedwater piping area. Adequate space is provided for the installation and support of the valves. Static or dynamic loads when operating or when subject to seismic events are considered.

The piping and valve arrangement and design analysis is performed in accordance with the guidelines in ASME Section III, Non-mandatory Appendix O, “Rules for Design of Safety Valve Installations.” (Reference 10.3-10)

Each MSSV is connected to a vent stack. The stacks are arranged and designed to prevent steam backflow from the transition piece and to minimize the backpressure on the valve outlet.

The vent stacks are designed and supported to withstand SSE loads. This is to prevent the vent stacks from being damaged and jeopardizing the performance of safety-related components.

10.3-6), Safety Class 2 and Seismic Category I.

C. Main Steam Relief Valve Block Valve

MSRVBVs with remote control are located upstream of each MSRVs and MSDVs facilitating isolation of leaking or stuck open MSRVs or MSDVs. MSRVBVs are closed manually from the main control room, and automatically close when steam line pressure reaches a predetermined set point.

10.3.2.3.4 Main Steam Isolation Valves and Main Steam Check Valves

The function of the MSIVs is to limit uncontrolled steam release from one SG in the event of a MSLB with a single active failure in order to:

- Limit the effect on the reactor core to within the specified fuel design limits.
- Limit containment pressure to a value less than the design pressure,

If the MSLB occurs upstream of MSIVs, ~~even if a single failure of this valve is assumed,~~ the broken side SG is isolated by the MSIVs on the main steam piping of the intact SGs or the MSIV/MSCV of the broken line. In case of a line break downstream of the MSIV, ~~even if a single failure of this valve is assumed,~~ MSIVs on the main steam piping of the both intact SGs and faulted SG would prevent the steam blowdown ~~through more than one SG~~. See Table 10.3.3-1 which shows failure of either train solenoid valve for MSIV actuation does not impair isolation function of MSIV.

MSIV consist of system medium actuated gate valve which uses valve inside pressure to close in each main steam line with instrumentation. These valves are located outside the containment in the main steam/feedwater piping area. The MSIVs are designed to fully close within 5 seconds after the receipt of following signals:

- Low main steam line pressure
- High-high containment pressure
- High main steam line pressure negative rate
- Manual actuation

Valve design parameters are provided in Table 10.3.2-2

10.3.2.3.5 Main Steam Bypass Isolation Valves

MSBIVs are installed in parallel to the MSIVs. MSBIVs are used to warm up main steam lines prior to start up when MSIVs are closed. The valves also equalize the pressure on either side of the MSIV to enable opening of the MSIV. Bypass valves are air-operated globe valves and are closed during normal plant operation. The valves are designed to close within 5 seconds automatically by the same signals for MSIVs.

Valve design parameters are provided in Table 10.3.2-2

10.3.2.3.6 Main Steam to Emergency Feedwater Pump Turbine

See Subsection 10.4.9, Emergency Feedwater System.

10.3.2.4 System Operation

10.3.2.4.1 Normal Operation

During startup, the main steam piping is heated by opening the MSBIV and thus controlling the steam flow. Main steam is not admitted to the main turbine until warmup of the main steam piping is accomplished. After warmup mode, secondary side no-load temperature and pressure are maintained automatically by the turbine bypass system which is maintained in the pressure control mode. When the reactor coolant temperature reaches 557°F (which is the no load temperature), the MSIVs are opened in a controlled manner. As the piping downstream of MSIVs is heated up, MSIVs are fully open and the MSBIVs are closed.

The MS/R 2nd reheat supply steam shutoff valve, control valve, bypass valve and warmup valve remain closed below 10% turbine load. With turbine load greater than 10%, heating steam is admitted by opening the warmup valve to the tube bundle.

During hot standby condition, the SG pressure is controlled by modulating TBVs and dumping steam to the condenser.

During plant cool down, decay and sensible heats are removed by dumping steam into the condenser via the TBVs. When the steam pressure falls below 125 psia, the steam dump is then stopped and cooldown is switched to the residual heat removal operation.

10.3.2.4.2 Emergency Operation

In the event that the plant must be shutdown due to accident or transient, the MSIVs with associated MSBIVs are closed. The MSDVs are used to remove the reactor decay heat and primary system sensible heat in order to cooldown the primary system to the conditions at which the residual heat removal system can perform the remaining cooldown function. If one of the MSDVs is unavailable, the respective safety valves associated with that main steam line provide overpressure protection. The remaining MSDVs are sufficient to cooldown the plant.

In the event of a design-basis accident, such as a main steam line break, the MSIVs with associated MSBIVs are automatically closed. In case the line break is downstream of the MSIV, ~~even if a single failure of this valve is assumed,~~ the MSIVs on the main steam piping of the both intact SGs and faulted SG would prevent the steam blowdown through more than one SG. If the MSLB occurs in the upstream of MSIVs, the broken side SG is isolated by the MSIVs on the main steam piping of the intact SGs or the MSIV/MSCV of the broken line.

10.3.2.4.3 Water (Steam) Hammer Prevention

Table 10.3.2-2 Main Steam System Valves (Sheet 1 of 3)

Main Steam Safety Valve

Number of valves per main steam line	6
Total number of valves	24
Relieving capacity per valve	884,000 (lb/hr) at design pressure
Relieving capacity per main steam line	~~5,302,500~~5,304,000 (lb/hr) at design pressure
Total relieving capacity	~~21,210,000~~21,216,000 (lb/hr) at design pressure
Valve type	Spring type
Valve size	6 (in)
Design pressure	1,185 (psig)
Design temperature	568 (°F)
Design code	ASME Section III, Class 2
	Seismic category I

Valve number	Set pressure (psig)	Relieving capacity (lb/hr)
MSS- SRV-509 (A,B,C,D)	1,185	884,000
MSS-SRV-510 (A,B,C,D)	1,215	906,000
MSS-SRV-511 (A,B,C,D)	1,244	928,000
MSS-SRV-512 (A,B,C,D)	1,244	928,000
MSS-SRV-513 (A,B,C,D)	1,244	928,000
MSS-SRV-514 (A,B,C,D)	1,244	928,000

Main Steam Relief Valve

Number per main steam line	1
Total number of valves	4
Valve size	6 (in)
Design capacity per valve	531,000 (lb/hr) at 1,150 (psig)
Total	2,121,000 (lb/hr) at 1,150 (psig)
Design pressure	1,185 (psig)
Design temperature	568 (°F)
Design code	ASME Section III, Class 2 Seismic category I
Actuator	Air-operated, modulating

Table 10.3.2-2 Main Steam System Valves (Sheet 3 of 3)

Main Steam Check Valves

Number per main steam line	1
Total number of valves	4
Valve size	32 (in)
Design pressure,	1,185 (psig)
Design temperature	568 (°F)
Design code	ASME Section III, Class 3 Seismic Category I
Actuator	-

Main steam bypass isolation valves

Number per main steam line	1
Total number of valves	4
Valve size	~~32~~4 (in)
Design pressure	1,185 (psig)
Design temperature	568 (°F)
Design code	ASME Section III, Class 2 Seismic Category I
Actuator	Air-operated, modulating

10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condensers

The main condenser functions to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system.

10.4.1.1 Design Basis

10.4.1.1.1 Safety Design Basis

The main condenser performs no safety-related function and therefore has no nuclear safety design basis.

10.4.1.1.2 Power Generation Design Basis

- The main condenser is designed to receive and condense the rated power exhaust steam flow from the low-pressure turbine and to perform as a reservoir for vents and drains from various components.
- The main condenser is also designed to receive and condense the turbine bypass steam up to 67.5 percent of plant rated steam flow, while condensing the residual low-pressure turbine steam flow. This condensing action is accomplished without exceeding the maximum allowable condenser backpressure for main turbine operation.
- At the normal operating water level, the condenser hotwell is designed for a five minute hold up time at rated condensate flow rate.
- The main condenser is designed to deaerate the condensate so that the dissolved oxygen in the condensate remains under 10 ppb during rated power operation.

10.4.1.2 System Description

The main condenser is part of the condensate system (CDS). The condensate system is described in Subsection 10.4.7 and shown in Figure 10.4.7-1 through 10.4.7-4. Classification of equipment and components is given in Section 3.2. Table 10.4.1-1 provides main condenser design data.

The main condenser is a three-shell, single-pass, single pressure, divided water boxes and rigidly supported unit. Each shell is located beneath its respective low-pressure turbine. The condenser is equipped with titanium tubes. The titanium material provides good corrosion and erosion resisting properties.

The condenser shells operate at the same pressure and temperature due to the equalizing pipe, which connects each condenser shell at neck area. Condensate is drawn from the hotwell of each condenser, and then flows through a single header to the suction of the condensate pumps.

The condenser shells are located below the turbine building operating floor and are rigidly supported on the turbine foundation. An expansion connection is provided between each low-pressure turbine exhaust opening and the steam inlet connections of the condenser. Four low-pressure feedwater heaters are located in the neck area of each condenser shell. Nozzles are provided at the bottom of condenser hotwell for instrumentation and control and leak detection connections.

10.4.1.2.1 System Operation

During normal power operation, exhaust steam from the low-pressure turbines is directed into the main condenser shells. The condenser also receives system flows from feedwater heater vents and drains and gland steam condenser drain.

The hotwell level controller provides automatic makeup or rejection of condensate to maintain a normal level in the condenser hotwells. On low level, the makeup control valves open and admit condensate to the hotwell from the condensate storage tank. On high-water level, the condensate reject control valves open to divert water from the condensate pump discharge to the condensate storage tank. This rejection automatically stops when the hotwell level reaches normal operating range.

Air inleakage and noncondensable gases contained in the turbine exhaust steam are collected in the condenser and removed by the main condenser air removal system. The main condenser evacuation system is discussed further in Subsection 10.4.2.

To protect the condenser shells and turbine outer casings from overpressurization, steam relief blowout diaphragms are provided in the low-pressure turbine outer casings. Pressure transmitters are provided on the condenser shells to detect the loss of the condenser vacuum. Pressure transmitters generate a turbine trip signal upon detecting the condenser pressure above its setpoint.

The main condenser is capable of accepting up to 67.5 percent of rated load main steam flow from the turbine bypass system. Operation of the turbine bypass system is discussed in Subsection 10.4.4.

In the event of a high condenser pressure or trip of all circulating water pumps, or trip of all condensate pumps, the turbine bypass valves are prohibited from opening.

Perforated distribution piping or baffle plates are installed to protect the condenser tubes, feedwater heaters located in the condenser neck, and other condenser components from turbine bypass steam or high-temperature drains entering the condenser shell.

The main condenser interfaces with the tube leak detection system as discussed in Subsection 9.3.2 to permit sampling of the condensate in the condenser hotwell. Should circulating water in-leakage occur, these provisions permit determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. This is performed by isolating the circulating water system from the affected water box. Plant power is reduced as necessary. The water box is then drained and the affected tubes are either repaired or plugged.

Table 10.4.1-1 Main Condenser Design Data

Condenser type		Horizontal, Radial Flow, Single Pressure, Single Pass, Surface Cooling Type
Number of Shell		3
Design operating pressure		2.6 in.-HgA
Heat transfer		9.90×10^{9} Btu/hr
Circulating water flow		1.28×10^{6} gpm
Circulating water inlet temperature		88.5°F
Circulating water outlet temperature		104°F
Circulating water temperature rise		15.5°F
Hotwell storage capacity		5 min. (holdup time)
Tube size		1 in. O.D. ~~22~~23 BWG
Shell pressure (design)		0 in.-HgA to 15 psig
Material	Shell	Carbon Steel
	Tube	Titanium
	Tube Sheet	Titanium Clad
	Water Box	Carbon Steel with rubber lining

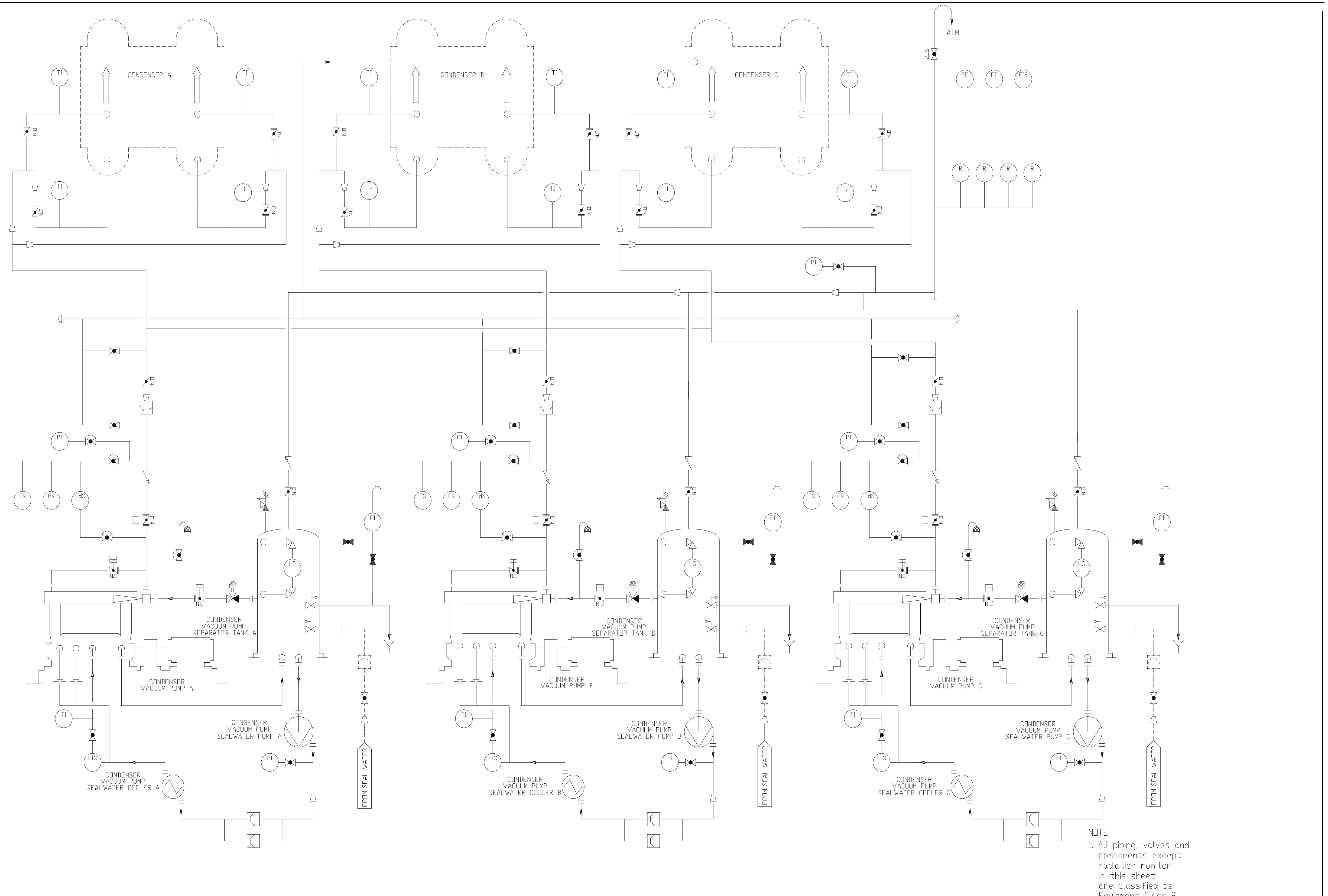


Figure 10.4.2-1 Main Condenser Evacuation System Piping and Instrumentation Diagram

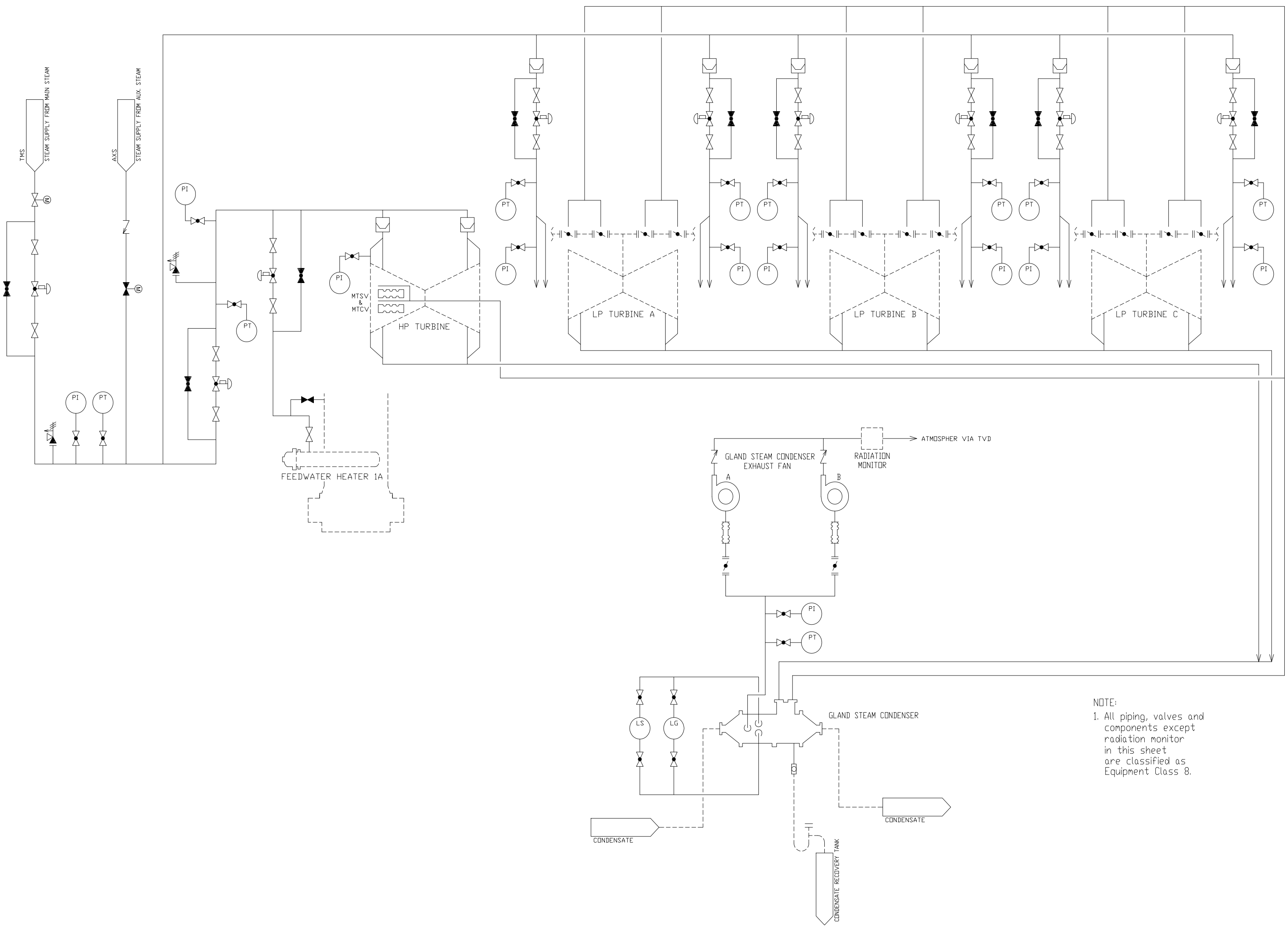


Figure 10.4.3-1 Gland Seal System Piping and Instrumental Diagram

10.4.4 Turbine Bypass System

The turbine bypass system (TBS) as described in this Subsection is part of the main steam system (MSS) and provides capability to flow the main steam from the steam generators (SG) to the main condenser bypassing the main turbine. This is done in a controlled manner to dissipate heat and to minimize transient effects on the reactor coolant system during startup, hot standby, cooldown and the generator step-load reduction.

10.4.4.1 Design Bases

10.4.4.1.1 Safety Design Bases

The TBS serves no safety-related function and thus has no nuclear safety design basis.

10.4.4.1.2 Non-safety Power Generation Design Bases

The following is a list of the non-safety power generation design bases:

- The TBS has the capacity to bypass 67.5 % of the rated power steam flow to the main condenser at full power operation.

- The TBS is designed to sustain a 100 % load rejection (electrical load), without generating a reactor trip, and without requiring actuation of the main steam relief valve (MSRV), main steam safety valve (MSSV) or pressurizer safety valve.

- The TBS is designed to bypass steam to the main condenser during plant shutdown to facilitate a manually controlled cooldown of the reactor coolant system to the point where the residual heat removal system can be placed in service for further cooldown.

- The TBS bypasses steam to the main condenser during plant startup

- The TBS is designed to follow rapid turbine load reductions greater than 10 % but less than 100 % without resulting in reactor trip.

10.4.4.2 System Description

10.4.4.2.1 General Description

The TBS is part of the MSS and is shown on Figures 10.3-2 and 10.3-3. The equipment and component classification and applicable codes and standards are provided in Section 3.2.

The TBS consists of a turbine bypass valve header tapped from the main steam equalization piping upstream of the main turbine stop valves, piping, valves and instrumentation. Two individual sub-headers per condenser shell are tapped from the bypass valve header. Lines with the TBVs are connected to these sub-headers. TBVs discharge to condenser shell(s) via two sub-headers per shell.

The load rejection controller prevents a large increase in the reactor coolant temperature following a large, sudden load decrease. The error signal is a difference between the lead-lag compensated selected T_{avg} and the selected Tref based on turbine inlet pressure and a difference between the nuclear power signal and the turbine inlet pressure with a rate-lag compensation.

Following a turbine trip, the load rejection controller is defeated and the turbine trip controller becomes active. The error signal is a difference between the lead-lag compensated T_{avg} and the no-load reference T_{avg}.

The pressure control mode is used at no-load operational mode. Pressure mode control is used to remove decay heat during plant startup and cooldown. The difference between the steam equalization piping pressure and a pressure set point is used to control the turbine bypass flow. The pressure set point is manually adjustable and is based on the desired reactor system coolant temperature.

10.4.4.4 Safety Evaluation

The TBS serves no safety function and has no safety design basis. There are no safety-related equipment/components in the vicinity of the TBS components. All high-energy lines of the TBS are located in the turbine building.

The failure of a TBS high-energy line will not disable the turbine speed control system.

The bypass valves fail closed upon loss of motive air power or electric signal. This is to prevent the possibility of the primary side of the plant from over cooling. In this case, MSRVs provide the controlled cooldown. In the unlikely event that one of the TBVs sticks wide open, the maximum steam flow through one valve at full load main steam pressure is less than the maximum permissible flow to limit a reactor transient.

The TBS is designed to bypass steam to the main condenser during normal plant shutdown. The system removes the residual heat and cools the reactor coolant system to a point where the RHR system is placed in service for further cooldown. Three TBVs with 13.~~4~~5 % of rated main steam flow of 20,200,000 lb/h at ~~the~~a valve inlet pressure of 792 psia~~774 psig~~ perform adequate decay heat removal to keep the cooldown rate of reactor coolant system at 50 deg.F/h during normal plant shutdown and thereby reduce the demands on systems important to safety in meeting GDC 34.

10.4.4.5 Inspection and Tests

Before the system is placed in service, all TBVs are tested for operability. The pipelines are hydrostatically tested to verify leak tightness. All piping and valves are accessible for inspection.

Additional description of inspection and tests is provided in Section 14.2.

10.4.4.6 Instrumentation Application

Instrumentation for the TBS is described in Section 7.7. Controls are provided in the main control room for the system operating mode selection. Pressure indication and the valve position indication are provided in the main control room.

Table 10.4.4-1 TBS Component Design Parameters

Turbine Bypass Valves

Number of valves	15
Capacity/valve (Requirement), (lb/hr)	909,000~~862,000~~
Total capacity (Requirement), (lb/hr)	13,635,000~~12,930,000~~
~~Total capacity (Available), (lb/hr)~~	~~13,604,000~~
Design Pressure (psig)	1,185
Design Temperature (°F)	568
Nominal valve size (inch)	10

power plant loading and design weather conditions. Design parameters for the major components are described in Table 10.4.5-1.

The CWS consists of [[two CTW assemblies]] which provide 100 percent cooling for normal power operation. [[Each CTW assembly contains two (2) back-to-back rows of ~~CTWs~~cells]]. The discharge piping from the circulating water pumps is [[headered together into an intake manifold concrete pipe]], as shown in Figure 10.4.5-1 The CWS supply and discharge piping to the three shell main condenser contains butterfly-type isolation valves.

[[Makeup water is provided by the raw water system to compensate for the CTW evaporation, drift and blowdown. The CTW water chemistry is controlled by the CWS/raw water system chemical treatment system.]] It should be noted that ~~two~~three non-essential service water (non-ESW) pumps are located in the turbine building, and two pumps are operated and take suction from the CWS piping in the turbine building. The non-ESW flows through the turbine component cooling water system (TCS) heat exchangers, and connects back to the main condenser outlet piping. In addition to the CWS flow, [[the CTW]] are sized to also cool the non-ESW flow. The non-ESW is described in Subsection 9.2.9.

10.4.5.2.2 Component Description

The circulating water system consists of the following major components:

- Circulating water pumps
- [[Cooling towers and CTW basins]]
- Main condenser
- Condenser tube cleaning equipment
- [[CTW make up water and blowdown system]]
- [[Chemical treatment system]]
- Instrumentation and controls

10.4.5.2.2.1 Circulating Water Pumps

The circulating water pumps [[(eight 12.5% capacity)]] are [[vertical pump, wet pit type, single-stage mixed flow pumps driven by direct drive electric motors. Each cooling tower basin contains four circulating water pumps that are arranged in parallel.]]

10.4.5.2.2.2 [[Cooling Towers]]

Table 10.4.5-1 Design Parameters for Major Components of Circulating Water System (Sheet 3 of 3) (see Note 1)

Piping and components design data	-
Design pressure/temperature, (psig/°F)	85/110
Material for Intake and discharge tunnel	Pre-stressed reinforced concrete with appropriate lining, if required by the CWS water chemistry.
Material for CWS above ground piping	ASTM A106, Grade B seamless carbon steel piping with lining ASTM A134, Grade C seam-welded carbon steel piping with lining
Type of CWS major valves	~~Motor-operated butterfly valves.~~ AWWA C504

Note:

1. Design parameters are dependent on site-specific conditions, and these values will change.

Table 10.4.6-1 Condensate Polishing System Design Parameters (Sheet 1 of 2)

Condensate polishing vessels

Number of vessels	3
Type	Vertical
~~Design pressure (psig) and Temperature (°F)~~	~~600 and 176~~
Design flow rate per vessel (gpm)	3,750
Maximum short term flow rate per vessel (gpm)	7,500 (Maximum flow occurs only for a short duration during the condenser tube leak operating period)
Materials of construction	Carbon steel with rubber lining

Prefilters

Number of vessels	3
Type	Non precoat type cartridge filters
~~Design pressure (psig) and Temperature (°F)~~	~~600 and 176~~
Design flow rate per filter (gpm)	3,750
Maximum short term flow rate per filter (gpm)	7,500 (Maximum flow occurs only for a short duration during the condenser tube leak operating period)
Materials of construction	Carbon steel with rubber lining

Table 10.4.6-1 Condensate Polishing System Design Parameters (Sheet 2 of 2)

Resin traps

Number of traps	3
Type	Basket
Materials of construction	Carbon steel with stainless steel strainer

Spent resin holding vessel

Number of vessels	1
Type	Vertical
~~Design pressure (psig)~~	~~100~~
~~Design temperature (°F)~~	~~176~~
Materials of construction	Carbon steel with rubber lining

Resin Mixing and Holding Vessel

Number of vessel	1
Type	Vertical
~~Design pressure (psig)~~	~~100~~
~~Design temperature (°F)~~	~~176~~
Materials of construction	Carbon steel with rubber lining

For a FLB upstream of the MFIV, the FWS is designed to prevent blowdown of any SG and also to maintain the emergency feedwater system (EFWS) in-flow to the SG.

The main feedwater check valve (MFCV), located between the MFIV and main feedwater regulationvalve (MFRV) in the main feedwater line to each SG, acts on reverse pressure differential. The MFCV is designed to withstand the forces encountered when closing after a FLB. The valves serve to prevent blowdown from more than one SG during a feedwater line break. During ~~normal or~~ upset or abnormal conditions, the function of these check valves is to prevent reverse flow from the SGs whenever the FWS is not in operation.

Main feedwater isolation is provided via the MFIVs. These valves are operated by separate solenoid valves with redundancy and independent class 1E power bus. The failure of one solenoid valve does not impair the isolation function of MFIV. MFIVs are designed to close automatically on main feedwater isolation signals within 5 seconds, an appropriate ECCS actuation signal, within the time established in Section 16.1.

10.4.7.1.3 Power Generation Design Basis

- The CFS is designed with the capability of automatically providing the required flow to the SGs during startup, shutdown, at power levels up to the rated power and during the plant design transients without interruption of operation or damage to equipment.
- Feedwater of uniform temperature is delivered to all SGs at any given power level. A continuous, steady feedwater flow is maintained at all loads.
- The system is able to accommodate ten percent step or five percent per minute ramp load changes without significant deviation from programmed water levels in the SGs or major effect on the feedwater system.
- The system has the capability of accommodating the necessary changes in feedwater flow to the SGs with the steam pressure increase resulting from a 100-percent load rejection.
- The plant is designed to operate at rated power with one condensate pump or feedwater booster/main feedwater pump assembly out of service
- With one feedwater heater string out of service, the plant is designed for operation at 70 percent of rated power.
- The feedwater and condensate pumps and pump control system are designed so that loss of one feedwater booster/main feedwater pump assembly or one condensate pump does not result in trip of the turbine-generator or reactor.
- The pumps and other system components are designed to avoid the need for an immediate trip of the condensate, feedwater booster/main feedwater pumps on low net positive suction heads.

MFIV is designed to be capable trip-closed within 5 seconds after receiving signals, such as ECCS actuation signal or high-high SG water level signal in any one of the SGs.

Redundant control and indication channels are provided for each of the isolation valves. Provisions are made for inservice inspection of the isolation valves.

Main Feedwater Regulation Valves:

The MFRVs are air-operated 16 inch size control valves with the purpose of controlling feedwater flow rate. The MFRV are designed to ASME Code Section III, Class 3 and seismic category I. The valve body is a globe design. Seats and trim are of an erosion resistant material. The design allows for removal and replacement of seats and other wearing parts. The MFRVs automatically maintain the water level in the SGs during operational modes. Positioning of the MFRV during normal operation is the function of an automatic SG water level control using a conventional three-element control scheme (feedwater flow, steam flow, SG water level).

MFRV is designed to close within 5 seconds after receiving signals, such as an ECCS actuation signal, high-high SG water level signal, ~~and~~ P-4 & low Tavg signal and high SG water level signal. Details of the three element control system are provided in Chapter 7.

Main Feedwater Check Valves:

Each main feedwater line includes the MFCV (18 inch size) installed outside containment. The valves are designed to ASME Code, Section III, Class 3, seismic category I. During normal and upset conditions, the MFCV prevents reverse flow from the SG whenever the feedwater pumps are tripped. In addition, the closure of the valves prevents more than one SG from blowing down in the event of a feedwater line break. The MFCV is designed to limit blowdown from the SG and to prevent water hammer due to sudden valve closure.

Main Feedwater Bypass Regulation Valves:

MFBRVs (6 inch size) are designed to ASME Code Section III, Class 3, seismic category I. MFBRVs are installed to bypass the MFRVs, and are utilized to adjust the main feedwater flow from approximately 3% up to 15% rated power. The main feedwater bypass control system is 3-element (feedwater flow, ΔT, SG water level) type control system.

The MFBRV is designed to close within 5 seconds after receiving signals, such as a ECCS actuation signal, ~~or,~~ a high-high SG water level signal and high SG water level signal.

Steam Generator Water Filling Control Valves:

SGWFCV is used from no load up to 3% by one element (SG water level only) controller. Details of the control are provided in Section 7.7.

Table 10.4.7-2 Major Component Design Parameters (Sheet 1 of 2)

Condensate pump	
Number	3
Type	Vertical, multistage, centrifugal
Driver	~~Synchronous~~ Induction ac motor
Rated flow (gpm)	12,500
Rater head (ft)	1,000
Rated power (HP)	4,500
Feedwater booster pump	
Number	4
Type	Centrifugal, horizontal
Driver	~~Synchronous~~ Induction ac motor (Main feedwater pump common use)
Rated flow (gpm)	16,700
Rater head (ft)	2,820 (the sum total with main feedwater pump)
Rated power (HP)	14,700 (the sum total with main feedwater pump)
Main feedwater pump	
Number	4
Type	Centrifugal, horizontal
Driver	~~Synchronous~~ Induction ac motor
Variable speed unit	Hydro coupling unit
Rated flow (gpm)	16,700
Rater head ()ft	2,820 (the sum total with feedwater booster pump)
Rated power (HP)	14,700 (the sum total with feedwater booster pump)
Low-pressure feedwater heater No.1	
Number	3
Type	Horizontal, single zone, shell and U-tube
Material, shell	Carbon steel
Material, tubes	Stainless steel
Heat duty (Btu/hr)	$7.4x10^8$
Low-pressure feedwater heater No.2	
Number	3
Type	Horizontal, two zone, shell and U-tube with drain cooler
Material, shell	Carbon steel
Material, tubes	Stainless steel
Heat duty (Btu/hr)	$4.6x10^8$
Low-pressure feedwater heater No.3	
Number	3
Type	Horizontal, two zone, shell and U-tube with drain cooler
Material, shell	Carbon steel
Material, tubes	Stainless steel
Heat duty (Btu/hr)	$4.4x10^8$

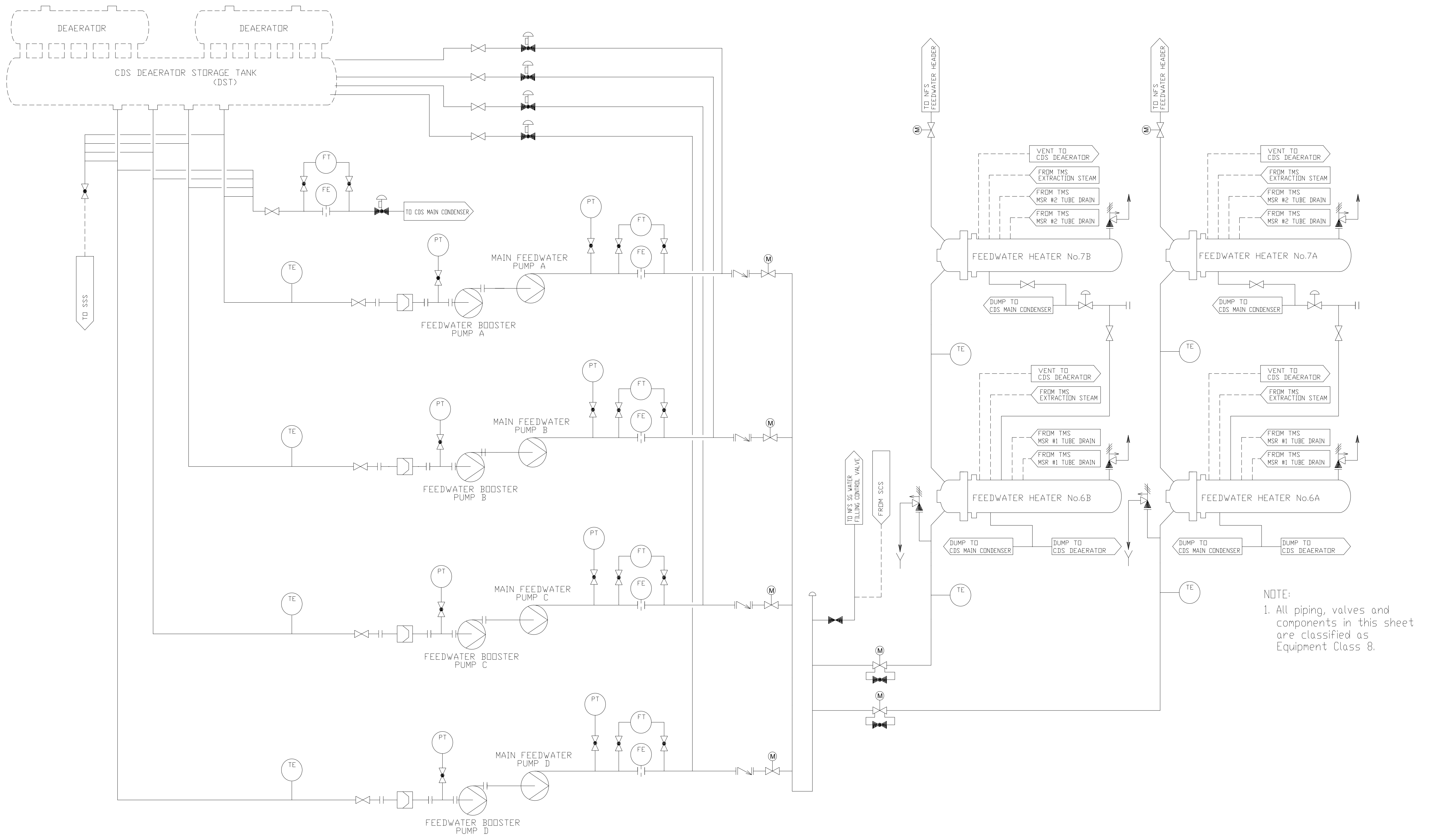


Figure 10.4.7-4 Condensate and Feedwater System Piping and Instrumentation Diagram (4/4)

10.4.8.2.1 General Description

The SGBDS flow diagrams are shown in Figures 10.4.8-1 and 10.4.8-2. Classification of equipment and components in the SGBDS is provided in Section 3.2.

The SGBDS equipment and piping are located in the containment, the reactor building, the auxiliary building and the turbine building (T/B).

The SGBDS consists of a flash tank, regenerative heat exchangers, non-regenerative coolers, filters, demineralizers, piping, valves and instrumentation. The flash tank, regenerative heat exchangers and non-regenerative coolers are provided to cool the blowdown water with heat recovery, while the filters and demineralizers are provided to purify the blowdown water.

One blowdown line per steam generator is provided. The blowdown from each steam generator flows independently to the flash tank. The blowdown water from the flash tank flows via one common line to regenerative heat exchangers and non regenerative coolers. Blowdown is split into two trains ahead of the heat exchangers. Common discharge from the coolers flows to the filters and demineralizers, where the flow is split into two trains. The purified water from the demineralizers flows to the condenser via a common discharge line.

The blowdown line from each steam generator is provided with two flow paths, a line for purifying blowdown water used during normal plant operation and a line for discharging the blowdown water to the [[WWS]] or the condenser used during startup and abnormal water chemistry conditions.

The US-APWR SG's utilize a "peripheral" blowdown system arrangement. In this arrangement, blowdown holes are drilled from approximately 7 inches below the secondary surface of the tubesheet and intersect with the peripheral groove on the secondary face of the tubesheet. This arrangement is shown as Figure 10.4.8-3 and facilitates effective sludge removal from the tubesheet. The blowdown from each steam generator is depressurized by a throttle valve located downstream of the isolation valves. The throttle valves can be manually adjusted to control the blowdown rate.

The depressurized blowdown water flows to the flash tank, where water and flashing vapor are separated. The vapor is diverted to deaerator and the water is transferred to regenerative and non-regenerative heat exchangers for further cooling. ~~During plant startup w~~When the pressure in the flash tank is low, the vapor is diverted to condenser. The condensate and feedwater system (CFS) provides the condensate in regenerative heat exchanger(s) to recover thermal energy.

The turbine component cooling water system (TCS) cools blowdown water in the non-regenerative heat exchanger to protect the demineralizer resin prior to purifying the blowdown water. The impurities from the cooled blowdown water are removed by the inlet filters, demineralizers and outlet strainers. SG blowdown demineralizers consist of two cation demineralizers and two mixed bed demineralizers. The purified water is returned to the condenser.

A local grab sample point which is provided downstream of each demineralizers, a radiation monitor downstream of demineralizers outlet strainers and a radiation monitor in the sample line measures impurities concentration and the radioactivity level in the blowdown water. In case of SG tube leakage and when abnormally high radiation level is detected, the blowdown lines are isolated and the blowdown water included in SGBDS is transferred to waste holdup tank in the liquid waste management system (LWMS). See Subsection 11.2.2 for details.

Regenerative heat exchangers and non-regenerative coolers consist of two- 50 % capacity trains. When blowdown flow rate is less than 0.5% Maximum Steaming Rate (MSR) at rated power, one regenerative heat exchanger and one non-regenerative cooler are in operation while the other regenerative heat exchanger and non-regenerative cooler can remain on standby or isolated for maintenance.

Demineralizers include two - 100 percent trains. Each demineralizer train includes cation demineralizer and mix bed demineralizer.

Furthermore filters are installed upstream these demineralizers and strainers are also installed downstream these demineralizers in this SG blowdown line.

During plant startup, blowdown rate is up to approximately 3% of MSR at rated power. In this mode, blowdown liquid flows to the [[WWS]] prior to discharge or to the condenser for processing in the Condensate Polishing System (CPS). During normal operation, blowdown rate is approximately 0.5 to 1% of MSR at rated power. At the 1% of MSR at rated power blowdown rate, both cooling trains are used.

With abnormal water chemistry, the flow of blowdown rate up to approximately 3% of MSR at rated power is directed to the [[WWS]] for processing prior to discharging to the environment.

A blowdown sample line from each steam generator is provided for sampling. A SG blowdown sample cooler is located in each of these lines for cooling blowdown liquid to reduce blowdown temperature suitable for the secondary water quality monitoring station. Cooled liquid flows to secondary water quality monitoring station, SG blowdown water radiation monitor and sample sink for taking grab samples if an SG tube leak occurs.

A secondary water quality monitoring station measures pH, specific conductivity, cation conductivity and sodium ion concentration in the secondary water continuously. Furthermore grab sampling points are provided for sampling secondary water and analyzing chloride and sulfate ion concentration in the secondary water.~~sulfate ion, chloride ion and sodium ion concentrations. The SG blowdown water radiation monitor is continuously utilized for SG tube leak detection.~~

Two isolation valves on each blowdown line are located in the main steam/feedwater piping area. The SG blowdown water is transferred through each SG blowdown line under normal operating and transient conditions. The isolation valves close automatically upon receipt of one of the following signals:

- High radiation signal from SG blowdown return water radiation monitor

- High-high radiation signal from SG blowdown water radiation monitor
- High-high radiation signal from condenser vacuum pump exhaust line radiation monitor
- Emergency feedwater pump automatic actuation signal
- ~~High water level in the blowdown flash tank~~
- ~~High pressure in the blowdown flash tank~~

In addition, the containment isolation valve closes automatically upon receipt of containment isolation signal.

The containment isolation valve in the blowdown sample line closes automatically upon receipt of one of the following signals:

- High radiation signal from SG blowdown return water radiation monitor
- High-high radiation signal from SG blowdown water radiation monitor
- High-high radiation signal from condenser vacuum pump exhaust line radiation monitor
- Emergency feedwater pump automatic actuation signal
- Containment isolation signal

10.4.8.2.2 System Operation

The various modes of operation are as follows:

10.4.8.2.2.1 Plant Startup

In this mode, the reactor is brought from cold shutdown to no-load power operating temperature and pressure.

The steam generator secondary side water chemistry is brought to operating specifications as rapidly as possible. High blowdown rates (up to 3% of MSR at rated power), are used to reduce the solids content in the steam generators. SG blowdown water is directed to the [[WWS]] or the condenser without passing through the SG blowdown demineralizers. The SG bulk water chemistry condition is maintained by discharging the SG blowdown water to [[WWS]]. To facilitate oxygen removal and enhance reduction of corrosion products in the system, chemical injection is provided in all volatile treatment (AVT) mode with higher levels of hydrazine.

As long as the SG blowdown water can be returned to the condenser, the SG blowdown water is transferred to the condenser for purification by the condensate polishers located in CPS .

10.4.8.2.2.2 Normal Operation

After the plant reaches rated power conditions, the hydrazine concentration is reduced to the concentration level for normal operation and the system remains in AVT mode. The normal blowdown flowrate varies from approximately 0.5 % to 1 % of MSR at rated power.

During normal operation, including SG tube leakage and condenser tube leakage within allowable limits and with low impurities, the CFS water chemistry is high in pH.

SG blowdown water is cooled in series of regenerative heat exchanger and non-regenerative cooler, purified by SG blowdown demineralizers and discharged to the condenser hotwell. After the initiation of purification in the SG blowdown demineralizers, all condensate water bypasses CPS.

The radioactive spent resins are transferred to solid waste management system (SWMS) for disposal, when SG tube leakage exceeds allowable limits and resins are non recyclable. During normal operation without SG tube leakage, non-radioactive spent resins are transferred to a non-radioactive spent resin holding vessel (SRHV) in CPS. These resins are shipped to an off-site facility for regeneration.

10.4.8.2.2.3 Plant Shutdown

In this mode, the reactor is brought from no-load power operating temperature and pressure to a cold shutdown.

High blowdown rates (up to 3% MSR at rated power) may be used to reduce the solids contents in the steam generators and maintain secondary water chemistry within allowable limits. The blowdown water is returned to the condenser or to the [[WWS.]]

10.4.8.2.2.4 Steam Generator Drain

The SGBDS is used to drain the steam generators. In this mode, the blowdown drain water is directed to the condenser or to the [[WWS]]. A small amount of drain water potentially remains in the steam generator blowdown piping and the drain water is discharge to the containment sump through a dedicated drain line which branches off from the lowest point of the blowdown line. The COL Applicant is to describe the nitrogen or equivalent system design for Steam Generator drain.

10.4.8.2.2.5 Abnormal Operation

(1) Condenser Tube leakage

The CPS goes into service and maintains the condensate water quality. SG blowdown water can be purified by the SG blowdown demineralizers to support purification of CPS or diverted directly to the condenser.

(2) SG blowdown lines isolation signals

of the SGBDS is designed to remain functional during and after a safe-shutdown earthquake.

- The safety-related components of the SGBDS are qualified to function in normal and accident environmental conditions. The environmental qualification program is described in Section 3.11.

- Section 3.2 provides quality group classification, design and fabrication codes, seismic category applicable to the SGBDS.

- Failure modes and effects analysis, as listed in Table 10.3.3-1, concludes that no single failure coincident with loss of offsite power compromises system's safety functions.

- High and moderate energy pipe break locations and its effects are discussed in Section 3.6.

- Coolant chemistry specifications to demonstrate compatibility with SG tube primary to secondary system pressure boundary material are addressed in Subsection 10.3.5. Preserving these specifications is accordingly able to control secondary water chemistry needed to maintain the integrity of the SG tube materials. Furthermore the description of the bases for the selected chemistry limit and secondary coolant chemistry program for steam generator blowdown sample are specified in Subsection 10.3.5.

10.4.8.4 Inspection and Tests

The SGBDS and components are tested in accordance with the plant procedures, during the initial testing and operation program. Since the SGBDS is in continuous use during normal plant operation and essential parameters are monitored, the satisfactory operation of the system and components demonstrate system operability. The safety-related components (piping and valves) are designed and located to permit preservice and inservice inspections to the extent practical.

Additional description of inspection and tests is provided in Chapter 14.

10.4.8.5 Instrumentation Applications

Pressure, flow, temperature and radiation instrumentation monitor and control the system operation.

High pressure and high water level in the blowdown flash tank closes the upstream flow control valve.

Flow elements located downstream of the isolation valves measure and ~~control~~indicate blowdown flow from each steam generator.

Temperature instrumentation monitors the temperature of the blowdown fluid upstream and downstream of the heat exchangers and the fluid temperature is limited below predetermined value into demineralizers. A high temperature signal upstream of SG blowdown demineralizers isolates the flow to the demineralizers. A setpoint temperature

Table 10.4.8-1 Steam Generator Blowdown System Major Component Design Parameters (Sheet 1of 3)

SG blowdown flash tank

Type	Vertical cylindrical	
Number of tanks	1	
Capacity (ft^3)	3~~0~~70	
Design flow rate (lb/hr)	202,000 (1% of MSR at rated power)	
Design pressure (psig)	300	
Design temperature (°F)	410	
Materials of construction	Stainless steel	

SG blowdown regenerative heat exchangers (per heat exchanger)

Type	Shell and tube	
Number of exchangers	2	
Design heat duty (Btu/hr)	17.4~~6~~x10^6	
Operating conditions	Tube side	Shell side
Fluid	SG blowdown water	Condensate
Operating temperature- In (°F)	375	129
- Out (°F)	158	365
Design flow rate (lb/hr)	78~~9~~.4x10^3	72~~3~~.7~~5~~x10^3
Design pressure (psig)	300	560
Design temperature (°F)	410	410
Materials of construction	Stainless steel	Carbon steel

SG blowdown non-regenerative coolers (per cooler)

Type	Shell and Tube	
Number of coolers	2	
Design heat duty (Btu/hr)	3.52~~7~~x10^6	
Operating conditions	Tube side	Shell side
Fluid	SG Blowdown Water	TCS
Operating temperature - In (°F)	158	100
- Out (°F)	113	109
Design flow rate (lb/hr)	78~~9~~.4x10^3	392~~7~~x10^3
Design pressure (psig)	300	200
Design temperature (°F)	200	200
Materials of construction	Stainless steel	Carbon steel

Table 10.4.8-1 Steam Generator Blowdown System Major Component Design Parameters (Sheet 2of 3)

SG blowdown demineralizers

Number of demineralizers	4 (two cation bed and two mixed bed)
Resin amount (ft^3)	230
Design flow rate (gpm)	3~~20~~16
Design pressure (psig)	300
Design temperature (°F)	200
Materials of construction	stainless steel

SG blowdown sample coolers

Type	Double tube
Number of coolers	4
Design heat duty (Btu/hr)	209×10^3

Operating conditions		Tube side	Shell side
Fluid		Blowdown water	CCW
Operating temperature	- In (°F)	557	100
	- Out (°F)	113	128
Design flow rate (lb/hr)		440	7,500
Design pressure (psig)		1185	200
Design temperature (°F)		568	200
Materials of construction		stainless steel	carbon steel

SG blowdown demineralizers inlet filters

Type		Vertical cylindrical, cartridge
Number of filters		2
Operating flow rate (gpm)		3~~20~~16
Operating temperature (°F)		113
Design pressure (psig)		300
Design temperature (°F)		200
0.8 micron particles retention (%)		98
Material of construction,	filter	Polyproplyene
	Body	Stainless steel

10.4.9 Emergency Feedwater System

The emergency feedwater system (EFWS) is designed to supply feedwater to the steam generators (SG) whenever the reactor coolant temperature is above 350°F and the feedwater system is not in operation. The EFWS is designed to remove reactor core decay heat and reactor coolant system (RCS) sensible heat through the SGs following transient conditions or postulated accidents such as a reactor trip, loss of main feedwater, main steam line breaks (MSLB) or feedwater line breaks (FLB), loss of offsite power (LOOP), small break loss of coolant accident (small break LOCA), station blackout (SBO), anticipated transient without scram (ATWS) and steam generator tube rupture (SGTR). The EFWS is not normally used during normal plant startup and normal plant cooldown.

The EFWS consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, piping, valves and associated instrumentation. The EFWS is an ASME Code, Section III (Reference 10.4-8), Classes 2 and 3, Seismic Category I, redundant system with Class 1E electric components as indicated in Table 3.2.2. The EFWS design meets the requirements of II.E.1.1 relating to reliability evaluation of the EFWS and II.E.1.2 of NUREG-0737 (Reference 10.4-13) regarding the automatic and manual initiation and flow rate indication of the EFWS.

The EFWS supplies feedwater to the SGs at a sufficient flowrate to meet the requirements for the transient conditions or postulated accidents and hot standby. Flowrate is controlled as necessary to maintain stable plant conditions by the motor-operated emergency feedwater control valves.~~.~~

10.4.9.1 Design Basis

The EFWS design bases to meet the safety-related functional requirements are provided below:

- The EFWS is designed to remain functional after a safe-shutdown earthquake (SSE). The essential portions of the EFWS components are designed to Seismic Category I requirements and are located inside the reactor building which is designed for seismic, wind and tornado effects. See Sections 3.2, 3.3, and 3.9.

- The EFWS components and piping have sufficient physical separation and shielding to protect against the effects of postulated missiles. Protection of the essential portions of the EFWS from the effects of internally and externally generated missiles is discussed in Section 3.5.

- The functional performance of the EFWS is not affected by environmental conditions, internal flood, pipe whip or jet impingement that may result from high or moderate energy piping breaks or cracks. The building where the EFWS components are located is designed for and provided with suitable flood protection during abnormally high water levels (adequate flood protection considering the probable maximum flood) to ensure functional capability. Flood protection is discussed in Section 3.4. Protection against the effects of pipe whip and jet impingement that may result from high energy piping breaks and moderate energy piping cracks is discussed Section 3.6. The environmental design of EFWS components is discussed in Section 3.11.

- A malfunction or single active failure of a system component or non-essential equipment does not reduce the performance capabilities of the EFWS. The EFWS and supporting systems ensure the required flow to the SGs in the event of a single active failure. The EFWS can perform all safety-related functions assuming a single active component failure in one train and a maintenance outage of one active component at on-line maintenance (OLM).

- The EFWS can utilize diverse power sources such that the system performance requirements are met with either power source (ac or dc). The EFWS satisfies the requirement that the pumps be powered by diverse power sources.

- Provisions are included to verify correct EFWS operation, to detect and control system leakage, and to isolate portions of the EFWS in case of excessive leakage or component malfunctions.

- The EFWS is designed with provisions to permit periodic inservice inspection and operational testing of the pumps and valves during normal plant condition.

- The EFWS is designed with I&C features to verify that the system is operating in the correct mode.

- The EFWS is designed to provide emergency feedwater (EFW) automatically for the removal of sensible heat and reactor core decay heat in order that there is no damage to the reactor core following a loss of main feedwater in order to bring the reactor core from a condition of full power to where the reactor coolant temperature is brought to the point at which the residual heat removal system (RHRS) may be placed in operation. The EFWS is automatically initiated by the EFW actuation signal such as LOOP signal, an ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or a low steam generator water level signal in any of the SGs. The automatic initiating circuits are powered from the emergency buses.

- The EFWS maintains the capability of the SGs to remove sensible heat and reactor core decay heat by converting the EFW to steam, which is then discharged to the atmosphere.

- The EFWS is capable of automatically initiating flow upon receipt of a EFW actuation signal. The system is also capable of manual actuation to provide protective action and for operational testing independent of the automatic signal. A single failure of the manual circuit does not result in loss of system function.

- The EFWS design is provided with the capability to automatically terminate EFW flow to a depressurized (faulty) SG and to automatically provide EFW to the intact SGs. The EFWS design is also capable of automatically terminating EFW flow to prevent overfilling of the SGs.

- The EFWS is designed such that in the unlikely event that the main control room (MCR) must be evacuated, the EFWS can be operated from the Remote Shutdown Console.

- The EFWS design meets the recommendations identified in NUREG-0611 (Reference 10.4-14).

- The EFWS design meets the provisions of TMI Action Plan Item II.E.1.2 of NUREG-0737 (Reference 10.4-13) regarding the automatic and manual initiation of the system, and 10 CFR 50.62(c)(1) (Reference 10.4-15) regarding the automatic initiation of the system on conditions indicative of an ATWS.

- The EFWS has the capability to permit operation at hot standby for 8 hours followed by 6 hours of cooldown to the RHR cut-in temperature from the MCR using only safety related equipment with a single active failure. The EFWS is designed with two EFW pits, both pits together providing a sufficient volume of water required for the emergency condition.

- The EFWS is designed with sufficient diversity to remain operable for a limited duration with neither offsite nor onsite ac power available. Turbine-driven pumps are designed to be available for SBO condition. Refer to Section 8.4 for the plant design to meet station blackout (SBO) requirements.

- Technical Specifications provide Limiting Condition for Operation and the surveillance testing requirements for EFWS to ensure continued system reliability during plant operation. See Chapter 16 for details.

- The EFWS is designed and constructed in accordance with ASME Code, Section III (Reference 10.4-8), Class 3 requirements up to the motor-operated EFW isolation valves (containment isolation valves). The containment isolation valves and the downstream piping to the feedwater system are safety class 2.

- The EFW pump main steam line steam isolation valves (containment isolation v~~c~~alve) in the steam supply lines and the steam piping upstream of the containment isolation valves are ASME Code, Section III (Reference 10.4-8), Class 2. The steam supply lines to the EFW pump turbine from ~~steam lines~~ the downstream of the containment isolation valves are designed and constructed in accordance with ASME Code, Section III (Reference 10.4-8), Class 3 requirements.

- The safety classifications are shown in the EFWS flow diagram shown in Figures 10.4.9-1 and 10.4.9-2. Codes and standards applicable to the EFWS and components are listed in Table 3.2-2.

- The principle emergency feedwater system materials are shown in Table 10.4.9-7.

- The recommendations of RGs 1.36 and 1.37 are applied during fabrication of the EFWS and preheat guidelines in ASME Code Section III, Appendix D, Article D-1000 for carbon steel are applied to the EFS component.

10.4.9.2 System Description

The EFWS flow diagram is shown in Figures 10.4.9-1 and 10.4.9-2. The system consists of two motor-driven pumps, two steam turbine-driven pumps, two EFW pits, and associated piping, valves, instruments and controls. The EFWS components are located in the reactor building. Table 10.4.9-1 provides data for the major components in the EFWS. Table 10.4.9-2 presents steam generator makeup flow requirements.

The EFWS is comprised of four 50% capacity pumps. Each EFW pump is sized to supply the feedwater flow required for removal of 50% of the decay heat from the reactor. The EFWS capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the RCS at an average temperature of approximately 50°F per hour. Main steam depressurization valves (MSDV) are used to relieve the steam produced by EFW during safe shutdown, following transient and accident conditions.

For a transient or accident condition, the EFW flow is delivered within 140 seconds of any automatic EFW actuation signal to at least two effective (intact) SGs.

The EFWS is designed with two 50% EFW pits, both pits together provide a sufficient volume of water required for the emergency condition.

The EFW flow is provided from the two EFW pits to the EFW pumps. The design of both EFW pits provides heat removal capability for a period of 14 hours. The total period of 14 hours consists of 8 hours at hot standby, and followed by a 6-hour cooldown of the primary system at an average rate of approximately 50°F per hour.

Each EFW pump discharge line connects with a tie line with a motor-operated isolation valve. During normal plant operation (at non-OLM), the discharge tie line isolation valves of each EFW pump discharge tie line are in the closed position to provide separation of four trains. During OLM, the tie line isolation valves of each EFW pump discharge tie line are kept in the open position. At OLM, all the discharge tie line isolation valves are required to be kept open to supply the specified flow rate of EFW to the SGs, assuming OLM of one EFW pump and the single failure of one of the three remaining EFW pumps.

The motor-operated EFW isolation valves and EFW control valves are provided in each EFW pump discharge line to close automatically to terminate the flow to the affected SG and continuously supply feedwater to the intact SG as discussed below:

A. Main feedwater line break

In the event of a FLB, the EFW line connected to that SG is automatically isolated by redundant motor-operated valves, which receive a low steam pressure signal to close. As a result, almost none of the EFW pump flow is lost by spilling out of the break. The logic is arranged so that only one EFW line can be automatically isolated.

B. Main steam line break

In the event of a MSLB, the SGs depressurize and the EFWS provides SG feedwater flow. In order to prevent excessive SG feedwater flow and pump runout, the motor-operated EFW control valve located in the EFW discharge line to each SG is provided with a pre-set open position. This position is adjusted and set during pre-operational testing. The line to the faulted SG is isolated automatically, as discussed above, for a rupture of a main feedwater line; isolation of the faulted SG and the termination of flow to the faulted SG limits the RCS cooldown and mass/energy release to the containment.

C. Steam generator tube rupture

Upon detection of a water level increase of the SG, the EFW isolation valves and EFW control valves are automatically closed.

The failure modes and effects analysis given in Table 10.4.9-4 demonstrates that required EFW flow is ensured to the SGs during postulated accident conditions with a single failure in the EFWS.

10.4.9.2.1 Description of Major Components

A description of the major components and features in the EFWS is as follows:

A. Emergency feedwater pumps

Each EFW pump is normally aligned to feed one SG. Each EFW pump takes suction from one of two EFW pits and the discharge flow is directed to one of the four SGs.

The EFW pump is designed to develop adequate head to supply the design flow of at least 400 gpm to each SG, when the SG pressure is equivalent to the set pressure of the first stage of the main steam safety valve (safety valve with lowest set pressure) plus 3% of accumulation and the pump discharge tie line is closed.

The maximum EFW pump flow is limited by the motor-operated EFW control valves which have a preset open position.

A mini flow line from the EFW pump discharge line to the EFW pit with a normally open valve and an orifice is provided to maintain minimum recirculation flow required for pump protection. The minimum flow line ensures a minimum recirculation flow for pump cooling whenever the pumps are running. A and B EFW pump shares their minimum flow line. C and D EFW pump also share~~s~~ their minimum flow line. Following the requirements in NRC IE Bulletin IEB 88-04, the minimum flow line ~~is given~~has sufficient capacity so that either of the pumps which share a minimum flow line does not ~~become~~ dead-head. A separate full flow line with a normally closed valve and an orifice allows pump testing during normal plant operation at the pump design flow rate without injection to the SGs. Both the mini flow line and full flow line are routed to the EFW pit by a common header.

Two motor-driven and two turbine-driven EFW pumps, with different power supplies are provided. Two motor-driven EFW pumps connect to each different safety ac bus to achieve the specific safety functions in case of off-site power loss; each bus is backed by a redundant emergency power source. Table 10.4.9-6 presents the power sources for

EFWS components.

The EFW pumps automatically start on receipt of LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low steam generator water level signal in any one of SGs.

B. Motor-driven (M/D) emergency feedwater pumps

Two of the four EFW pumps are horizontal, centrifugal pumps driven by electric motors which are supplied with power from independent, Class 1E Safety ac bus. Each motor-driven pump has a capacity of 450 gpm. The capacity of each motor-driven pump is based on the required flow of 400 gpm to SG and 50 gpm through miniflow line. The design parameters of the pump and the motor are provided in Table 10.4.9-1.

C. Turbine-driven (T/D) emergency feedwater pumps

Two of the four EFW pumps are turbine-driven providing diversity of motive pumping power. The pump is a horizontal, centrifugal unit with a capacity of 550 gpm. The capacity of each turbine-driven pump is based on the required flow of 400 gpm to SG and 150 gpm through miniflow line.

The steam supply line to each T/D EFW pump turbine is connected to main steam lines from two SGs. Steam supply piping to the turbine driver for the A-EFW pump is taken from the two main steam lines (A-main steam Line and B-main steam Line) and the steam supply piping to the turbine driver for the D-EFW pump is taken from the two main steam lines (C-main steam Line and D-main steam Line). The steam supply connection is made upstream of the MSIVs. The motor-operated isolation valve and a check valve are provided in each of these steam lines to the EFW pump turbine. The check valves prevent blowdown from an intact SG into a faulted SG. The MOV provides isolation of these lines in case of a SGTR. The steam line to each T/D-EFW pump is also provided with a normally closed motor-operated EFW pump actuation valve. Opening of this valve starts the T/D EFW pumps. The steam discharge from the T/D-EFW pumps is routed to the atmosphere. The design parameters of the pump and the motor are provided in Table 10.4.9-1.

D. Emergency feedwater pits

Two 50% EFW pits are provided. The EFW pits are completely enclosed stainless steel lined structures that do not contain any operating equipment. All components inside the pit are also constructed of stainless steel. No foreign materials intrusion is anticipated. An access hatch located above the 100 % water level is available for inspections of pit interior areas. The EFW pits are filled with clean demineralized water. Filtration is not required. Both EFW pits together contain the minimum water volume required for maintaining the plant at hot standby condition for 8 hours and performing plant cooldown for 6 hours until the RHRS can start to operate. The inside dimensions of each pit is approximately 28 feet long, approximately ~~42~~43 feet wide and approximately 35 feet deep~~depth~~. With the minimum pit level ~~at~~of 92.5% ~~approximately 26 feet~~ during normal plant condition, the volume of water in each pit available for the EFW is 186,200 gallon. With two pits, each ~~pit~~ with a capacity of 241,000~~204,850~~ gallons at the water levels from 0 to 100%, it is sufficient to perform hot standby and plant cooldown until the RHRS starts to perform heat removal. And also each pit has adequate capacity ~~for~~from the pit low level alarm setpoint to allow at least 20 minutes for operator action in accordance with the additional short-term recommendation “Primary EFW Water Source Low Level Alarm,” of recommendations of NUREG-0611 and NUREG-0635.

The makeup line routed from the demineralized water storage tank to the EFW pit is used for initial water fill of the EFW pits and to provide makeup water to maintain the water level in the EFW pits during normal plant operation. The demineralized water storage tank provides a backup source for EFWS. Due to a sufficient volume of water in the EFW pits for safe shutdown ~~of~~by keeping the plant at hot standby for 8 hours and performing plant cooldown to RHR entry condition for 6 hours after accident or transient, this backup supply is not required to be safety-related. The manual valves from the demineralized water storage tank to the EFW pumps are normally closed. If the water level of both EFW pits reaches low-low water level after an accident or transient without stabilizing at MODE 4 condition, the manual isolation valve will be opened by an operator. Before opening the isolation valve, the operator will verify that the storage tank has adequate water level to keep sufficient NPSH of the EFW pumps.

The common suction line from each EFW pit is connected by a tie line with two normally closed manual valves. When the two EFW pumps taking suction from the same pit are not available (OLM of one EFW pump and the single failure of other EFW pump), the tie line connections to EFW pits need to be established. In this case, to prevent depletion of the water source from one pit, the tie line valves at the EFW pit outlet are required to be opened within about 8 hours after starting EFW pumps to perform continuous feedwater supply to the intact SGs. The design parameters of the EFW pit are provided in Table 10.4.9-1.

Because the EFW pits have the water supplied directly from demineralized water ~~condensate~~ storage tank without deaerating and the inventory water of the pit has direct contact with atmosphere, the dissolved oxygen level of the pit inventory is not zero, however, because the design temperature of the EFW pit is ~~105~~150 Deg F, which is determined to exceed assumed maximum operating temperature of the EFWS, ~~the~~ stress corrosion cracking would not occur in such low temperature condition even if the level of dissolved oxygen is high, therefore, the EFW pits have adequate integrity.

Sampling of the EFW pits is performed monthly, and turbidity is ensured to be not over 1 ppm. Any deviation is corrected by utilizing ~~bleed~~feed and ~~feed~~bleed method. Demineralized water from the demineralized water storage tank (make-up water source) is used for feeding the water inventory. Complete inspections with the pits drained will be performed periodically per the ISI program.

E. Emergency feedwater control valves

The normally open motor-operated globe control valves are provided in the EFW pump discharge lines to each SG for controlling the EFW flow. The control valve pre-set open position is established during pre-operational testing to limit the maximum flow during steam line break accidents. These flow control valves also provide isolation function of the EFW to the faulty SG.

The motor-operated valves are normally-open and verified whether they are in pre-set open position at startup of the EFW pump on receipt of open check signal such as LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low steam generator water level signal in any one of the SGs. The motor-operated valves are also closed on receipt of such signal as high SG water level or low main steam line pressure. The design parameters of these valves are provided in Table 10.4.9-1.

F. Emergency feedwater isolation valves

The motor-operated gate isolation valves are provided in the EFW lines routed from the EFW pump to each SG for isolation of the EFW to the faulty SG.

The motor-operated valves are normally-open and verified whether they are in fully open position at startup of the EFW pump on receipt of a open check signal such as LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low SG water level signal in any one of SGs. The motor-operated valves are also closed on receipt of such signal as high SG water level or low main steam line pressure. The design parameters of these valves are provided in Table 10.4.9-1.

G. Turbine-driven EFW pump main steam-line steam isolation valves

The EFW pump turbine steam isolation valves are normally open dc motor-operated gate valves. One valve is provided in each line from the SG that provides steam to the EFW pump turbine. These valves are containment isolation valves. They are closed if required to terminate a leak or break or ~~if~~when the EFW pump actuation valve requires maintenance. The valves are operated from the MCR. The design parameters of these valves are provided in Table 10.4.9-1.

H. Turbine-driven EFW pump actuation valves

There are ~~two~~four normally closed EFW pump actuation dc motor-operated valves. ~~One~~Two valve ~~is~~are provided for each steam supply line to EFW pump turbine from two main steam lines~~in the common line that provides steam to the EFW pump turbine~~. The valves automatically open upon receiving EFW pumps actuation signal. The design parameters of these valves are provided in Table 10.4.9-1.

10.4.9.2.2 System Operation

A. Operation During Normal Plant Operation

(a) Plant Startup

The EFWS is not used during plant startup.

(b) Normal Plant Operation

The EFWS is not in operation during normal plant operation and is in standby mode. The EFW pit water level is maintained at, or above, the minimum required inventory to ensure adequate RCS heat removal and cooldown in the event of the failure of the feedwater system.

The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally opened.

The EFW isolation valves and control valves in the M/D and T/D EFW pumps discharge paths to the SGs are normally opened.

(c) Plant Shutdown

increases, the flow required from the EFWS decreases because the safety injection flow removes more decay heat from the core. Eventually, for large break LOCAs, safety injection flow removes the decay heat and no EFW is required from the EFWS.

(d) Feedwater Line Break (FLB)

F~~M~~LB is a postulated accident assuming that the main feedwater piping between the SG and the main feedwater check valve ruptures during normal plant operation. At this time, water inventory in the faulted SG is depleted, and main feedwater and EFW spills out of the break, resulting in reduction of heat removal in the secondary side and leading to temperature increase of the RCS. Hence, it is necessary to isolate the faulted SG and supply EFW to the intact SGs.

The EFW pump automatically starts following FLB. Upon detection of a main steam pressure decrease in the faulted loop, the faulted loop is automatically isolated and continuous EFW is supplied to the intact SGs.

(e) Main Steam Line Break (MSLB)

The most limiting condition resulting from a spectrum of MSLB is a double-ended rupture of a main steam line, occurring at zero power. The accident results in a severe cooldown transient. The EFWS is expected to provide the maximum SG feedwater flow rate because that makes the cooldown more severe until the affected SG is isolated. The EFWS is required to limit its feed flow to the SGs, especially to the faulted SG. The flow from the EFW line to the faulted SG is isolated automatically as described in the FLB accident analysis. The EFW supply function is not needed during the mitigation of the MSLB accident~~, but is needed only for cooldown up to the RHR system initiation~~.

(f) Station Blackout (SBO)

A SBO results in the loss of normal offsite and emergency onsite ac power sources. The M/D-EFW pumps are inoperable because there is no ac power. Both T/D EFW pumps are available because of the dc power supplied by class 1E batteries with two hours capacities. EFW flow control is also available because the EFW flow control valves are powered by dc power which is available from class 1E batteries. In addition, at least within one hour after the SBO occurrence, one unit of the AAC-GTG is started, and by the operation of one unit of emergency feedwater pump (turbine-driven) area air handling units, the integrity of one unit of T/D EFW pump is ensured. The AAC-GTGs minimize the potential for common cause failures with the Class 1E GTG as discussed in Section 8.4.1.3. From the above, because the AAC GTGs are available during SBO event, in accordance with the generic recommendations of NUREG-0611 and NUREG-0635 Generic Short Term Recommendation No. 5 (GS-5), the EFWS is capable of providing required EFW flow for at least two hours from one T/D-EFW pump. After starting the operation of the AAC-GTG, charging to the Class 1E batteries ~~are~~is resumed, therefore, the turbine-driven EFW pump is able to continue to operate after two hours of the SBO and is independent of any ac power source.

(g) Anticipated Transient Without Scram (ATWS)

The acceptance criteria for an ATWS is to provide adequate heat removal such that the maximum RCS pressure is limited to less than the emergency stress limit. For this event, the EFWS is actuated by the DAS (diverse actuation system).

(h) Steam Generator Tube Rupture (SGTR)

The SGTR is a postulated accident that assumes that, a SGTR and the reactor coolant flows to the secondary side of the SG. The EFW pump automatically starts on receipt of an ECCS actuation signal. Upon detection of a water level increase in the faulted SG, the EFW isolation valve on piping to the faulted~~all~~ SG is automatically closed. When all pumps start and operate without failure, the SG water level is verified in all SGs. If there is no potential for decrease in SG level, the pump is stopped depending on the condition. The emergency operating procedures provide additional details for operator actions during the accident conditions.

A summary of system performance for various accident conditions is provided in Table 10.4.9-3. The table includes flows to both the faulted and intact SGs. Comparing these data with those in Table 10.4.9-2, it is seen that minimum flow requirements for the intact SGs are satisfied under all failure modes.

C. Water Hammer Prevention

The following items are identified as water hammer prevention and mitigation measures in EFWS.

- Automatic initiation of EFW flow following a loss of main feedwater flow to prevent draining of the SG feedring in accordance with NUREG-0927
- Implementation of EFW pipe refill flow limits to minimize steam-water entrainment and subsequent formation of water slug~~. I~~in accordance with BTP 10-2
- Detection of a high temperature main feedwater back leakage from an EFW check valve which becomes the cause of water hammer

The Combined License Applicant is to provide operating and maintenance procedures in accordance with NUREG-0927 and a milestone schedule for implementation of the procedure. The procedures should address:

- Prevention of rapid valve motion
- Introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Introduction of steam or heated water that can flash into water-filled lines and components
- Introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

water level of the EFW pits during normal plant condition, monitor water level following an accident and annunciate abnormal water level. A non-safety water level sensor diverse from the safety-related water level sensors are installed in each EFW pit. The EFW discharge line temperature upstream of the EFW flow control valves is monitored. A high temperature alarm in the MCR is an indication of the back leakage of the check valve, requiring operator action.

Safety-related display instrumentation related to the EFWS is discussed in Section 7.5. Information indicative of the readiness of the EFWS prior to operation and the status of active components during system operation is displayed for the operator in the MCR and at the remote shutdown console. See Section 7.4 for details. The indication and controls provided for the EFWS are summarized in Table 10.4.9-5.

Sections 7.3 and 7.5 describe instrumentation design details for actuating, monitoring and controlling operation of the EFWS, including alarm and system actuation.

10.4.9.3 Safety Evaluation

The EFWS components, instrumentation, and power supplies are sized and designed with sufficient redundancy to maintain the safety-related functions of the system under all credible transient and accident conditions. The combination of turbine-driven pumps and motor-driven pumps provides a diversity of motive power sources to assure delivery of feedwater under all transient and accident conditions.

The EFWS and supporting systems are designed to provide the required flow to the SGs with a LOOP, assuming a single active component failure in one train and a maintenance outage of one active component at On-Line Maintenance.

The EFWS, with two Seismic Category I EFW pits, provides a means of pumping sufficient feedwater to remove the core decay heat following a loss of main feedwater event as well as to cool down the RCS to a temperature of 350°F at which point the RHRS can operate. A minimum of 186,200 gallons of water in each of the EFW pits is sufficient to supply the required water volume to SGs under all conditions. The basis for 186,200 gallons of water in each of the EFW pits is as follows:

Decay heat during hot standby (8 Hours) and cooldown (6 Hours)	: 225,900 gallon
Sensible heat to be removed from hot standby condition to start of residual heat removal	: 62,300 gallon
RCP heat input removal (one pump operation for 14 hours)	: 31,800 gallon
SG water level restore volume (from hot standby to cooldown condition)	: 52,400 gallon
Total required EFW volume	: 372,400 gallon
Required EFW volume per pit	: 186,200 gallon
Total required EFW volume with 10 % margin	: 409,700 gallon

Total required EFW volume with 10 % margin per pit : 204,850 gallon

During a loss of main feedwater event, the SG water level lowers, and the heat removal capability of the SGs reduces, then the reactor coolant temperature increases and expansion of the reactor coolant results in a pressurizer water level increase. The EFW pump capacity is based on providing sufficient feedwater supply to prevent the reactor coolant discharge from the pressurizer safety valve even when only two EFW pumps and two SGs are available due to single failure of one EFW pump and one SG failure.

The EFW pump capacity established above also satisfies the required feedwater flow to SG to prevent the reactor coolant release from the pressurizer safety valve with loss of the main feedwater due to the main feedwater line break.

The EFW Pump capacities and start times (maximum of 140 seconds for M/D pump and 60 seconds for T/D pump) are established such that the above objectives are met and the EFW Pumps can deliver the required flow for all conditions as given in Tables 10.4.9-2 and 10.4.9-3. Pump head is sufficient to establish the minimum necessary flow rate against the SG pressure corresponding to the first stage main steam safety valve set pressure plus 3% accumulation pressure. The maximum time to start the electric motors and the steam turbines which drive the EFW pumps are chosen so that sufficient flow can be supplied to SGs during the feedwater line break event which can result in reactor core damage. See Section 15.2 for details.

~~With the low-low water level in the EFW pits the available net positive suction head (NPSH) to the M/D EFW pumps is 97 feet, while the maximum required NPSH is 73 feet providing adequate margin. The available net positive suction head (NPSH) to the T/D EFW pumps is 100 feet, while the maximum required NPSH is 76 feet providing sufficient margin.~~

The EFWS is designed to reduce the probability of steam binding. When ~~a~~ back leakage from an EFW check valve occurs, high temperature water from the main feedwater line will ~~retain~~remain around the check valve~~, and then steam voids may be formed due to the back leakage, which may become the cause of~~ resulting in the formation of steam voids which could lead to water hammer. ~~When~~As the leakage continues, the voids may reach~~es~~ ~~into~~the EFW~~-~~ pump casing and ~~into~~ suction line ~~and therefore, steam binding may occur which would make~~ creating the possibility for steam binding which would render the EFW pump inoperable. ~~-~~To avoid water hammer and steam binding ~~to~~of the EFW pump, monitoring ~~of~~ the EFW discharge line temperature upstream of the EFW check valve~~s~~ will provide~~s~~ early detection of back leakage~~, which requires~~ and allow for prompt corrective action. ~~These are~~This is especially important during ~~OLM because the pump discharge tie line is opened and the possibilities of all EFW becoming inoperable increases~~on-line maintenance that requires the pump discharge tie line to be open increasing the possibility for all EFW pumps to become inoperable. ~~In the case leakage from the EFW check valve is detected, restoration is~~ Should leakage be detected when the tie line is open, prompt restoration will be performed by the following procedure.

1. Isolate the relevant line using the EFW isolation valve (EFS-MOV-019), EFW pump outlet manual isolation valve (EFS-VLV-013) and EFW pump discharge cross-connect line isolation valve (EFS-MOV-014).

Table 10.4.9-1 Emergency Feedwater System Component Design Parameters (Sheet 1 of ~~3~~4)

Motor-Driven Emergency Feedwater Pump

Number of pumps	2
Type	Horizontal, centrifugal
Capacity (gpm)	450 (including minimum flowrate)
Total dynamic head (ft)	~~3,200~~3,120
Minimum flow rate (gpm)	50
~~NPSH available maximum operating flow (ft)~~	~~approx. 97~~
~~Material~~ ~~Impeller~~ ~~Casing~~ ~~Shaft~~	 ~~Stainless steel~~ ~~Stainless steel~~ ~~Stainless steel~~
Equipment Class	3
Design Code	ASME Section III, Class 3
Seismic Category	I
~~Motor~~ ~~Horse Power~~ ~~rpm~~ ~~Power Supply~~ ~~Design Code~~	 ~~800~~ ~~3,600~~ ~~6.9kV, 60Hz, 3 phase, Class 1E~~ ~~NEMA~~

Table 10.4.9-1 Emergency Feedwater System Component Design Parameters (Sheet 2 of ~~3~~4)

Turbine-Driven Emergency Feedwater Pump

Number of pumps	2
Type	Horizontal, centrifugal
Capacity (gpm)	550 (including minimum flowrate)
Total Dymamic Head (feet)	~~3,200~~3,120
Minimum Flowrate (gpm)	150
~~NPSH available maximum operating flow (ft)~~	~~approx. 100~~
~~Material~~ ~~Impeller~~ ~~Casing~~ ~~Shaft~~	 ~~Stainless steel~~ ~~Stainless steel~~ ~~Stainless steel~~
Equipment Class	3
Design Code	ASME Section III, Class 3
Seismic Category	I

Table 10.4.9-1 Emergency Feedwater System Component Design Parameters (Sheet 3 of ~~3~~4)

Emergency Feedwater Pit (per pit)

Number of pits	2
Pit inside dimensions, L(ft)×W(ft)×H(ft)	28 x ~~42~~43 x 35
Capacity (gallons)	241,000
Required volume (gallons)	186,200
Seismic Category	I

Emergency Feedwater Control Valves

Number of valves	4
Type	Globe valve
Size (inches)	3
Design pressure (psig)	2,135
Design temperature (°F)	~~105~~150
Material	Carbon steel
Design Code	ASME Section III, Class 3
Equipment Class	3
Seismic Category	I

Emergency Feedwater Isolation Valves

Number of valves	4
Type	Gate valve
Size (inch)	3
Design pressure (psig)	2,135
Design temperature (°F)	~~568~~150
Material	Carbon steel
Design Code	ASME Section III, Class 2~~3~~
Equipment Class	2
Seismic Category	I

Table 10.4.9-1 Emergency Feedwater System Component Design Parameters (Sheet 4 of 4)

Turbine-driven EFW pump main steam-line steam isolation valves

Number of valves	4
Type	Gate valve
Size (inches)	8
Design pressure (psig)	1,185
Design temperature (°F)	568
Material	Carbon steel
Design Code	ASME Section III, Class 2
Equipment Class	2
Seismic Category	1

Turbine-driven EFW pump actuation valves

Number of valves	4
Type	Globe valve
Size (inches)	8
Design pressure (psig)	1,185
Design temperature (°F)	568
Material	Carbon steel
Design Code	ASME Section III, Class 3
Equipment Class	3
Seismic Category	I

Table 10.4.9-2 Steam Generator Makeup Flow Requirement

Event	Flow requirement
Feedwater line break	705 (gpm) to 2 SGs.

Table 10.4.9-3 Emergency Feedwater Flow Information for Various Postulated Events

Events		Number of Pumps in Operation	Minimum Flow to the Intact Steam Generator
Loss of main feedwater	non-OLM	All four pumps are running	1600 gpm for 4 SGs.
		3 of 4 emergency feedwater pumps are running	1200 gpm for 3 SGs.
	during OLM, the EFW pump discharge tie line is opened	3 of 4 emergency feedwater pumps are running	1200 gpm for 4 SGs.
		2 of 4 emergency feedwater pumps are running	800 gpm for 4 SGs.
Feedwater line break	non-OLM	All four pumps are running	1200 gpm for 3 SGs.
		3 of 4 emergency feedwater pumps are running (failure of pump in malfunctioning train)	1200 gpm for 3 SGs.
		3 of 4 emergency feedwater pumps are running (failure of pump in intact train)	800 gpm for 2 SGs.
	during OLM, the EFW pump discharge tie line is opened	3 of 4 emergency feedwater pumps are running	1200 gpm for 3 SGs. (The EFW line for the faulty SG is automatically closed)
		2 of 4 emergency feedwater pumps are running	800 gpm for 3 SGs. (The EFW line for the faulty SG is automatically closed)
Plant Cooldown	non-OLM	All four pumps are running	1600 gpm[(Note)] for 4 SGs.
		3 of 4 emergency feedwater pumps are running	1200 gpm[(Note)] for 3 SGs.
	during OLM, the EFW pump discharge tie line is opened	3 of 4 emergency feedwater pumps are running	1200 gpm[(Note)] for 4 SGs.
		2 of 4 emergency feedwater pumps are running	800 gpm[(Note)] for 4 SGs.

(Note) Initial flow rates to SGs are shown. The flow rates will be decreased to prevent SG overfilling and to keep SG water level.

Table 10.4.9-4 Emergency Feedwater System Failure Modes and Effects Analysis (FMEA) (Sheet 2 of 3)

Components	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection	Remarks
2. T/D-EFW pump actuation valve EFS-MOV-103A, B,C,D normally closed, dc MOVs	Failure to open on demand	Loss of Non-emergency AC power Loss of Nominal Feedwater Small Break Loss of Coolant Accident Safe Shutdown	**No effect on safety-related function since:** The three remaining EFW pumps are sufficient for providing EFW to three SGs.	Valve information: Valve open/close position indication in MCR	The left columns describe the non-OLM case where the EFWS is separated into four trains (EFW pump discharge tie line is closed). For OLM: No effect on safety-related function since at least two pumps are available to operate and at least three SGs can be supplied with EFW by opening the EFW pump discharge tie line during all modes of plant operation assuming that one pump is not available due to maintenance.
		Feedwater System Pipe Break Steam Generator Tube Rupture Safe Shutdown	**No effect on safety-related function since:** Each EFW line is provided with redundant isolation valves that automatically close to isolate the affected SG. This permits the EFW supply to be provided to the three intact SGs by three pumps following the event. In addition, two pumps are available for supplying EFW to the two intact SGs assuming one T/D-EFW pump actuation valve failure.	Valve information: Valve open/close position indication in MCR	
3. EFW control valve EFS-MOV-017A, B,C,D normally opened, dc MOVs	Failure to close on demand	Inadvertent secondary depressurization Feedwater System Pipe Break Steam System Piping Failure Steam Generator Tube Rupture	**No effect on safety-related function since:** The series of this control valves and the isolation valves (EFS-MOV-019A,B,C,D) with redundancy can stop EFW supply to the affected SG (automatically closes upon receipt of signals).	Valve information: Valve open/close position indication in MCR	

Table 10.4.9-4 Emergency Feedwater System Failure Modes and Effects Analysis (FMEA) (Sheet 3 of 3)

Components	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection	Remarks
4. **EFW isolation valve** EFS-MOV-019A, B,C,D normally opened, dc MOVs These valves are normally positioned to limit the maximum EFW flow	Failure to close on demand	Inadvertent secondary depressurization Feedwater System Pipe Break Steam System Piping Failure Steam Generator Tube Rupture	**No effect on safety-related function since:** The series of this isolation valves and the control valves (EFS-MOV-017A,B,C,D) with redundancy can stop EFW supply to the affected SG (automatically closes upon receipt of signals).	Valve information: Valve open/close position indication in MCR	
5. **T/D-EFW pump main steam line steam isolation valve** EFS-MOV-101A, B,C,D normally opened, dc MOVs	Failure to close on demand	Loss of Coolant Accident	**No effect on safety-related function since:** Containment boundary remains intact with redundancy provided by this valve, SGs and main steam lines.	Valve information: Valve open/close position indication in MCR	
	Failure to close on demand	Steam Generator Tube Rupture	**No effect on safety-related function since:** Isolation of affected SG is achieved by redundant T/D-EFW pump actuation valves (EFS-MOV-103A,B,C,D).	Valve information: Valve open/close position indication in MCR	

Table 10.4.9-5 Emergency Feedwater System Summary of Indication and Controls

Indication

Parameter	Main control board and remote shutdown console
EFW pump discharge pressure	Y
EFW pump discharge flow	Y
EFW isolation/control valve position	Y
Turbine-driven EFW pump main steam line isolation valve position	Y
Turbine-driven EFW pump actuation valve position	Y
EFW pump discharge tie line isolation valve position	Y
EFW pit level (safety and non-safety)	Y
Motor-driven EFW pump run status	Y
EFW pit level alarms	Y
EFW pump discharge line temperature alarm	Y

Note: Y = Yes

Control

Motor-driven EFW pumps	Y
Turbine-driven EFW pumps	Y
EFW isolation/control valves	Y

Note: Y = Yes

Table 10.4.9-6 Emergency Feedwater System Electric Power Sources

Component	Component Number	Electric Train
A-Emergency feedwater pump (turbine-driven, for inside electrical components)	EFS-MPP-001A	Class 1E dc bus “A”
B-Emergency feedwater pump (motor-driven)	EFS-MPP-001B	Class 1E ac bus “B”
C-Emergency feedwater pump (motor-driven)	EFS-MPP-001C	Class 1E ac bus “C”
D-Emergency feedwater pump (turbine-driven, for inside electrical components)	EFS-MPP-001D	Class 1E dc bus “D”
A-Emergency feedwater control valve	EFS-MOV-017A	Class 1E dc bus “A”
B-Emergency feedwater control valve	EFS-MOV-017B	Class 1E dc bus “B”
C-Emergency feedwater control valve	EFS-MOV-017C	Class 1E dc bus “C”
D-Emergency feedwater control valve	EFS-MOV-017D	Class 1E dc bus “D”
A-Emergency feedwater isolation valve	EFS-MOV-019A	Class 1E dc bus “B”
B-Emergency feedwater isolation valve	EFS-MOV-019B	Class 1E dc bus “A”
C-Emergency feedwater isolation valve	EFS-MOV-019C	Class 1E dc bus “D”
D-Emergency feedwater isolation valve	EFS-MOV-019D	Class 1E dc bus “C”
A-Emergency feedwater pump (turbine-driven) actuation valve on A-steam supply line	EFS-MOV-103A	Class 1E dc bus “A”
A-Emergency feedwater pump (turbine-driven) actuation valve on B-steam supply line	EFS-MOV-103B	Class 1E dc bus “A”
A-Emergency feedwater pump (turbine-driven) A-main steam line steam isolation valve	EFS-MOV-101A	Class 1E dc bus “~~A~~D”
A-Emergency feedwater pump (turbine-driven) B-main steam line steam isolation valve	EFS-MOV-101B	Class 1E dc bus “D”
D-Emergency feedwater pump (turbine-driven) actuation valve on C-steam supply line	EFS-MOV-103C	Class 1E dc bus “D”
D-Emergency feedwater pump (turbine-driven) actuation valve on D-steam supply line	EFS-MOV-103D	Class 1E dc bus “D”

D-Emergency feedwater pump (turbine-driven) C-main steam line steam isolation valve	EFS-MOV-101C	Class 1E dc bus “A”
D-Emergency feedwater pump (turbine-driven) D-main steam line steam isolation valve	EFS-MOV-101D	Class 1E dc bus “~~D~~A”

10.4.11 Auxiliary Steam Supply System

The auxiliary steam supply system (ASSS) supplies the auxiliary steam required for plant use during plant startup, shutdown, and normal operation. Steam is supplied from either an auxiliary boiler or main steam.

10.4.11.1 Design Bases

10.4.11.1.1 Safety Design Bases

The ASSS has no safety-related function and therefore has no nuclear safety design basis.

10.4.11.1.2 Power Generation Design Basis

The ASSS has the functions as shown below:

- During plant normal operation, the ASSS supplies auxiliary steam to the components of primary system or HVAC system by taking the part of the main steam. Then the system transfers the condensed water from these components to the condenser to use the water as the steam again.
- During plant startup, shutdown and plant regular inspections, main steam is not available; the auxiliary steam from the auxiliary boiler is supplied to the components of the primary system, HVAC system and secondary system. The condensed water from the primary or HVAC system components are collected to the auxiliary boiler and used as auxiliary steam again, and the auxiliary steam sent to the secondary system is collected to the condenser or Condensate and Feedwater System (CFS).
- The auxiliary steam drain monitors the leakage of the radioactive materials from the boric acid evaporator to the condensed water of the ASSS.

10.4.11.2 System Description

10.4.11.2.1 General Description

The ~~conceptual flow~~ ASSS piping and instrumentation diagram is shown in Figure 10.4.11-1.

The system includes a control valve to reduce the main steam pressure, auxiliary boiler package, auxiliary steam drain tank, auxiliary steam drain pump, auxiliary steam drain monitor, auxiliary steam drain monitor heat exchanger, condensed water piping and other components.

The components served by the system are categorized into two groups.

Group I components are shown below. For operation as required during startup, shutdown, plant regular inspection and normal operation, the auxiliary steam from the auxiliary boiler or main steam is supplied to the components continuously or intermittently. Condensed water from these components is collected in the auxiliary steam drain tank

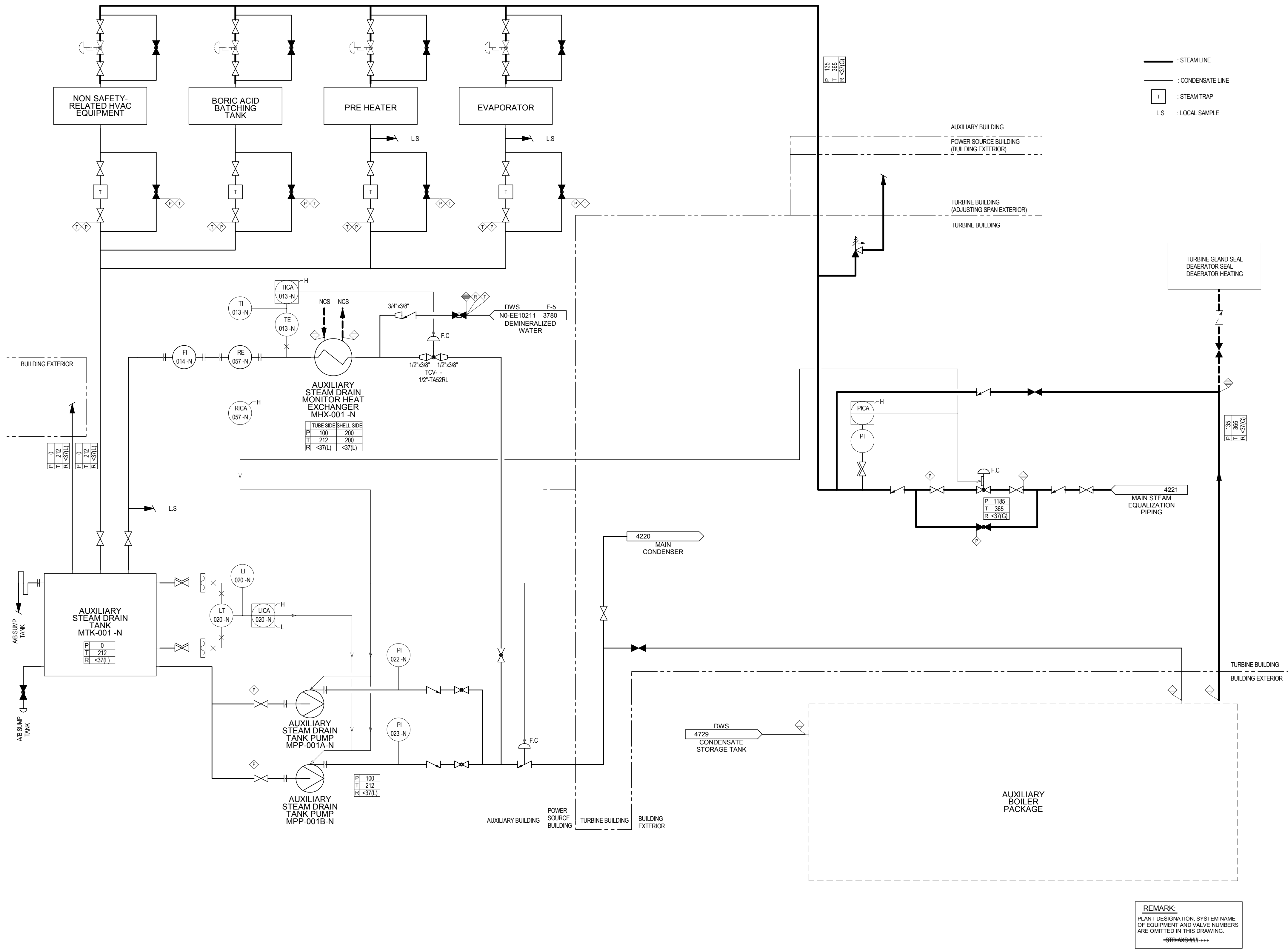


Figure 10.4.-11-1 Auxiliary Steam Supply System Piping and Instrumentation Diagram

Chapter 11

US-APWR DCD Chapter 11 Rev. 2, Tracking Report Rev. 7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
11.2-20	Subsection 11.2.6	Correction (typographical correction) Change the description of reference 11.2-6.
11.2-31	Table11.2-9	Correction (editorial correction) Delete duplicable row.
11.2-36,38	Table 11.2-11	Other Correct the design basis liquid effluent(maximum fuel defect) of some nuclides
11.2-41,42	Table 11.2-13	Other Correct the design basis liquid concentration(maximum release)
11.3-14	Subsection 11.3.3.2	Correction(editorial correction) Correct from “7.12E-12” to “7.25E-12” in the equation 11.3-4.
11.3-16	Subsection 11.3.7	Design progress Change the COL applicant as per design progress.
11.4-12	Subsection 11.4.2.3	Correction (typographical correction) Change the reference No.
11.4-21	Subsection 11.4.9	Correction (editorial correction) Delete reference 11.4-22 for duplication.

COL 11.2(6) The COL applicant is to provide piping and instrumentation diagrams (P&IDs).

COL 11.2(7) The COL Applicant is responsible for identifying the implementation milestones for the coatings program used in the LWMS. The coatings program addresses RG 1.54 Revision 1, recognizing that more recent standards may be used if referenced in DCD Section 11.2.

COL 11.2(8) The COL applicant is to describe mobile/portable LWMS connections that are considered non-radioactive but later may become radioactive through contact or contamination with radioactive systems (i.e., a non-radioactive system becomes contaminated due to leakage, valving errors, or other operating conditions in the radioactive systems), and operational procedures of the mobile/portable LWMS connections. The COL applicant is to prepare a plan to develop and use operating procedures so that the guidance and information in Inspection and Enforcement (IE) Bulletin 80-10 (Ref. 11.4-25) is followed.

~~11.2.5~~11.2.6 References

11.2-1 Standards for Protection Against Radiation, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20, December 2002.

11.2-2 Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix I .

11.2-3 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, Regulatory Guide 1.143, Rev. 2, November 2001.

11.2-4 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix A.

11.2-5 Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents-Nuclear Power Reactors, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50.34a.

11.2-6 Liquid Radioactive Waste Processing for Light Water Reactor Plants, ANSI/ANS 55.6, American National Standards Institute/American Nuclear Society, July 1993 (~~Revision~~ Reaffirmed 1999).

11.2-7 Minimization of Contamination, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20.1406.

Table 11.2-9 Input Parameters for the PWR-GALE Code (Sheets 1 of 2)

Design Parameter	Design Value
Core thermal power (MWt)	4,451
Reactor coolant mass(lb)	6.46E+05
Reactor coolant letdown flow rate (gpm)	180
CVCS cation demineralizer flow rate (gpm)	7
Number of SGs	4
Total main steam flow rate (lb/hr)	2.02E+07
Seccondary coolant mass in SG (lb)	1.35E+05
Total SG blowdown flow rate (lb/hr)	1.554E+05
Blowdown treatment method	0
Regeneration time of condensate polishing system	N/A
Fraction of feedwater through the condensate polishing system	0
Reactor coolant leak rate to the containment for noble gas (1/d)(1)	0.0002
Decontamination factor for detergent waste	1.0
Shim Bleed	-
Shim bleed flow rate (gpd)	2,875
Decontamination factor for I	5.0E+03
Decontamination factor for Cs and Rb	2.0E+03
Decontamination factor for others	1.0E+05
Collection time (days)	20
Process and discharge time (days)	2
Fraction of waste to be discharged	1.0
Coolant Drain	-
Coolant drainage flow rate (gpd)	900
Fraction of reactor coolant activity	0.1
Decontamination factor for I	5.0E+03
Decontamination factor for Cs and Rb	2.0E+03
Decontamination factor for others	1.0E+05
Collection time (days)	20
Process and discharge time (days)	2
Fraction of waste to be discharged	1.0
Dirty Waste	
Dirty drainageflow rate (gpd)	2,023
Fraction of reactor coolant activity	0.18
Decontamination factor for I	1.0E+05
Decontamination factor for Cs and Rb	2.0E+02
~~Decontamination factor for Cs and Rb~~	~~2.0E+02~~
Decontamination factor for others	1.0E+04
Collection time (days)	5
Process and discharge time (days)	0
Fraction of waste to be discharged	1.0
Blowdown Waste	
Fraction of the blowdown stream processed	1.0
Decontamination factor for I	1.0E+02
Decontamination factor for Cs and Rb	1.0E+02
Decontamination factor for others	1.0E+03
Collection time	N/A
Process and discharge time	N/A
Fraction of waste to be discharged	0
Regenerant Waste	N/A

Notes:

1: This value is based on 10 gpd of leakage inside containment (to containment sump)(see ~~table~~Table11.2-2) and reactor coolant mass.

Table 11.2-11 Liquid Releases with Maximum Defined Fuel Defects (Ci/yr)

(Sheets 1 of 2)

Isotope	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Detergent Waste	TOTAL Releases (1)
Corrosion and Activation Products						
Na-24	0.00000	0.00029	0.00002	0.00031	0.00000	3.20E-04
P-32	0.00000	0.00000	0.00000	0.00000	0.00018	1.80E-04
Cr-51	0.00000	0.00008	0.00000	0.00008	0.00470	4.78E-03
Mn-54	0.00000	0.00004	0.00000	0.00004	0.00380	3.84E-03
Fe-55	0.00000	0.00003	0.00000	0.00003	0.00720	7.23E-03
Fe-59	0.00000	0.00001	0.00000	0.00001	0.00220	2.21E-03
Co-58	0.00000	0.00012	0.00000	0.00012	0.00790	8.02E-03
Co-60	0.00000	0.00001	0.00000	0.00001	0.01400	1.40E-02
Ni-63	0.00000	0.00000	0.00000	0.00000	0.00170	1.70E-03
Zn-65	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
W-187	0.00000	0.00002	0.00000	0.00002	0.00000	2.06E-05
Np-239	0.00000	0.00003	0.00000	0.00003	0.00000	~~3.10E-05~~ 3.09E-05
Fission Products						
Rb-88	0.00000	0.03849	0.00000	0.03849	0.00000	3.97E-02
Sr-89	0.00000	0.00000	0.00000	0.00000	0.00009	9.00E-05
Sr-90	0.00000	0.00000	0.00000	0.00000	0.00001	1.00E-05
Sr-91	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Y-91m	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Y-91	0.00000	0.00000	0.00000	0.00000	0.00008	8.00E-05
Y-93	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Zr-95	0.00000	0.00001	0.00000	0.00001	0.00110	1.11E-03
Nb-95	0.00000	0.00002	0.00000	0.00002	0.00190	1.92E-03
Mo-99	0.00000	0.01333	0.00000	0.01333	0.00006	1.38E-02
Tc-99m	0.00000	0.00527	0.00000	0.00527	0.00000	5.44E-03
Ru-103	0.00000	0.00001	0.00000	0.00001	0.00029	3.00E-04
Rh-103m	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
Ru-106	0.00000	0.00001	0.00000	0.00001	0.00890	8.91E-03
Rh-106	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
Ag-110m	0.00000	0.00000	0.00000	0.00000	0.00120	1.20E-03
Ag-110	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Sb-124	0.00000	0.00000	0.00000	0.00000	0.00043	4.30E-04
Te-129m	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Te-129	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Te-131m	0.00000	0.00033	0.00000	0.00033	0.00000	3.40E-04
Te-131	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
I-131	~~0.00113~~ 0.02891	~~0.00056~~ 0.01445	0.00000	~~0.00169~~ 0.04336	0.00160	~~3.34E-03~~ 4.63E-02
Te-132	0.00000	0.00526	0.00000	0.00526	0.00000	5.43E-03
I-132	0.00000	0.00015	0.00015	0.00030	0.00000	~~3.10E-04~~ 3.09E-04
I-133	0.00163	0.00327	0.00491	0.00981	0.00000	1.01E-02
I-134	0.00000	0.00005	0.00000	0.00005	0.00000	5.16E-05
Cs-134	0.73457	1.83643	0.00000	2.57100	0.01100	2.66E+00
I-135	0.00000	0.00083	0.00125	0.00208	0.00000	2.15E-03
Cs-136	0.12019	0.44873	0.00000	0.56892	0.00037	5.87E-01
Cs-137	0.43698	1.16528	0.00000	1.60226	0.01600	1.67E+00

Notes:

Table 11.2-11 Liquid Releases with Maximum Defined Fuel Defects (Ci/yr)

(Sheets 2 of 2)

Isotope	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Detergent Waste	TOTAL Releases (1)
Ba-137m	0.20917	0.00000	0.00000	0.20917	0.00000	2.16E-01
Ba-140	0.00000	0.00010	0.00000	0.00010	0.00091	1.01E-03
La-140	0.00000	0.00002	0.00000	0.00002	0.00000	2.06E-05
Ce-141	0.00000	0.00000	0.00000	0.00000	0.00023	2.30E-04
Ce-143	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Pr-143	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Ce-144	0.00000	0.00001	0.00000	0.00001	0.00390	3.91E-03
Pr-144	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
TOTAL (except H-3)	**~~1.50367~~ 1.53145**	**~~3.51883~~ 3.53272**	**0.00633**	**~~5.02883~~ 5.07050**	**0.08975**	**~~5.28E+00~~ 5.32E+00**
H-3 release						1.60E+03

Notes:
1. The release totals include an adjustment of 0.16 Ci/yr added by the PWR-GALE Code to account for AOOs.
2. An entry of 0.00000 indicates that the value is less than 1.0E-5 Ci/yr.

Table 11.2-13 Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 (Maximum Releases) (Sheets 1 of 2)

Isotope(1)	Discharge Concentration (µCi/ml) (2)	Effluent Concentration Limit (µCi/ml) (3)	Fraction of Concentration Limit
Na-24	1.56E-11	5.00E-05	~~3.12E-07~~3.11E-07
P-32	8.77E-12	9.00E-06	9.74E-07
Cr-51	2.33E-10	5.00E-04	4.66E-07
Mn-54	1.87E-10	3.00E-05	6.24E-06
Fe-55	3.52E-10	1.00E-04	3.52E-06
Fe-59	1.08E-10	1.00E-05	1.08E-05
Co-58	3.91E-10	2.00E-05	1.95E-05
Co-60	6.82E-10	3.00E-06	2.27E-04
Ni-63	8.28E-11	1.00E-04	8.28E-07
Zn-65	~~5.03E-13~~5.02E-13	5.00E-06	~~1.01E-07~~1.00E-07
W-187	~~1.01E-12~~1.00E-12	3.00E-05	3.35E-08
Np-239	1.51E-12	2.00E-05	7.54E-08
Rb-88	1.93E-09	4.00E-04	4.84E-06
Sr-89	4.38E-12	8.00E-06	5.48E-07
Sr-90	4.87E-13	5.00E-07	9.74E-07
Sr-91	0.00E+00	2.00E-05	0.00E+00
Y-91m	0.00E+00	2.00E-03	0.00E+00
Y-91	3.90E-12	8.00E-06	4.87E-07
Y-93	0.00E+00	2.00E-05	0.00E+00
Zr-95	5.41E-11	2.00E-05	2.70E-06
Nb-95	9.35E-11	3.00E-05	3.12E-06
Mo-99	6.73E-10	2.00E-05	3.36E-05
Tc-99m	2.65E-10	1.00E-03	2.65E-07
Ru-103	1.46E-11	3.00E-05	4.88E-07
Rh-103m	~~5.03E-13~~5.02E-13	6.00E-03	~~8.38E-11~~8.37E-11
Ru-106	4.34E-10	3.00E-06	1.45E-04
Ag-110m	5.84E-11	6.00E-06	9.74E-06
Sb-124	2.09E-11	7.00E-06	2.99E-06
Te-129m	0.00E+00	7.00E-06	0.00E+00
Te-129	0.00E+00	4.00E-04	0.00E+00
Te-131m	1.66E-11	8.00E-06	2.07E-06
Te-131	0.00E+00	8.00E-05	0.00E+00
I-131	~~1.63E-10~~2.26E-09	1.00E-06	~~1.63E-04~~2.26E-03
Te-132	2.64E-10	9.00E-06	2.94E-05
I-132	1.51E-11	1.00E-04	1.51E-07
I-133	4.93E-10	7.00E-06	7.04E-05
I-134	2.51E-12	4.00E-04	6.28E-09
Cs-134	1.30E-07	9.00E-07	1.44E-01

Notes:
1. Rh-106,Ag-110,Ba-137m are not included in Table 2 of 10 CFR 20 Appendix B. Therefore, these nuclides are excluded from the calculation of the discharge concentration.
2. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 12,900 gpm dilution flow.
3. 10 CFR 20 Appendix B, Table 2

Table 11.2-13 Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 (Maximum Releases) (Sheets 2 of 2)

Isotope [(1)]	Discharge Concentration (μCi/ml) [(2)]	Effluent Concentration Limit (μCi/ml) [(3)]	Fraction of Concentration Limit
I-135	~~1.05E-10~~1.04E-10	3.00E-05	3.48E-06
Cs-136	2.86E-08	6.00E-06	4.77E-03
Cs-137	8.13E-08	1.00E-06	8.13E-02
Ba-140	4.93E-11	8.00E-06	6.17E-06
La-140	~~1.01E-12~~1.00E-12	9.00E-06	1.12E-07
Ce-141	1.12E-11	3.00E-05	3.73E-07
Ce-143	0.00E+00	2.00E-05	0.00E+00
Pr-143	0.00E+00	2.00E-05	0.00E+00
Ce-144	1.90E-10	3.00E-06	6.35E-05
Pr-144	~~5.03E-13~~5.02E-13	6.00E-04	~~8.38E-10~~8.37E-10
H-3	7.79E-05	1.00E-03	7.79E-02
TOTAL			**~~3.09E-01~~3.11E-01**

Notes:
1. Rh-106,Ag-110,Ba-137m are not included in Table 2 of 10 CFR 20 Appendix B. Therefore, these nuclides are excluded from the calculation of the discharge concentration.
2. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 12,900 gpm dilution flow.
3. 10 CFR 20 Appendix B, Table 2

A_i = Reactor coolant activity for noble gas nuclide i that is equal to 300 μCi/g dose equivalent Xe-133 as stated in the technical specifications (μCi/g)

A_i' = Realistic reactor coolant activity for noble gas nuclide i that is calculated by the PWR-GALE Code (μCi/g)

Dose to total body due to charcoal bed leak is calculated using the equation based on BTP 11-5(Ref. 11.3-18) shown below.

$$D = \sum_i K_i \cdot Q_i \cdot (X/Q) \cdot (1E+12\,\mathrm{pCi/Ci}) \cdot (7.12E-12\,\mathrm{yr}^2/\mathrm{event}-\mathrm{s}) \qquad \text{Eq. 11.3-4}$$

7.25E-12

where:

D = Dose to total body (mrem)

Q_i = Annual effluent of noble gas nuclide i (Ci/yr)

K_i = Dose factor of nuclide i for total body given as DFBi in Table B-1 of Regulatory Guide 1.109(mrem-m3/pCi/yr)

(X/Q)= The short-tem atmospheric dispersion factor at EAB specified in Chapter 2, Subsection 2.3.4(s/m^3)

The parameters for the calculation and the results are tabulated in Table 11.3-4. The dose at EAB is 2 mrem in case of a charcoal bed leak, which is lower than the dose in case of a waste gas surge tank leak and is less than the criterion of 100 mrem specified in BTP 11-5 (Ref. 11.3-18).

11.3.3.3 Offsite Dose Calculation Manual

The offsite dose calculation manual contains site-specific requirements. To assist the preparation of DCD and COLA, the applicant will use the NEI's generic templates for the offsite dose calculation manual, including radiological effluent technical specifications and the radiological effluent monitoring program. These templates were issued to Nuclear Regulatory Commission (NRC) for evaluation.

11.3.4 Ventilation System

The ventilation system is designed in accordance with the requirements of Guidance of RG 1.140 (Ref. 11.3-16). and is described in Chapter 9, Section 9.4. Chapter 12, Section 12.3 describes the radiation activity levels in the ventilation system.

The HVAC ventilation flow provides dilution for the GWMS release in the vent stack. The discharge isolation valve, located downstream of the discharge radiation monitor, closes on a low ventilation system exhaust flow rate to minimize the potential for the

11.3.7 Combined License Information

COL 11.3(1) Deleted

COL 11.3(2) Deleted

COL 11.3(3) ~~The COL applicant is to provide a discussion of the onsite vent stack design parameters and release point specific characteristics.~~The COL applicant is to provide a discussion of the onsite vent stack released point height.

COL 11.3(4) Deleted

COL 11.3(5) Deleted

COL 11.3(6) The COL applicant is to calculate doses to members of the public following the guidance of RG 1.109(Ref. 11.3-19) and RG 1.111(Ref. 11.3-22), and compare the doses due to the gaseous effluents with the numerical design objectives of 10 CFR 50, Appendix I (Ref. 11.3-3) and compliance with requirements of 10 CFR 20.1302(Ref. 11.3-24), 40 CFR 190(Ref. 11.3-25).

COL 11.3(7) Deleted

COL 11.3(8) The COL applicant is to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.

COL 11.3(9) The COL applicant is to provide piping and instrumentation diagrams (P&IDs).

11.3.8 References

11.3-1 Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code), NUREG-0017, Rev. 1, April 1985.

11.3-2 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, Regulatory Guide 1.143, Rev. 2, November 2001.

11.3-3 Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, NRC regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix I.

11.3-4 Standards for Protection Against Radiation, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20, December 2002.

the waste is also monitored during the filling operation to insure meeting disposal requirements for the licensed land disposal facility. Each container of processed waste is classified as Class A, B, or C waste using a site-specific 10 CFR 61 (Ref. 11.4-~~22~~6) Waste Form, in compliance with the site-specific process control program.

Some of the dry active waste is only slightly contaminated and permits contact handling. The SWMS design does not include compaction equipment or drum dryer equipment but provides the flexibility for the site-specific utilities to add compaction equipment or to adopt contract services from specialized facilities. The COL applicant is required to provide information on the adoption of compaction equipment.

Storage for packaged radioactive wastes is provided in a shielded area. The storage area is conveniently located next to the truck bay at an elevation of 3' 7" in the A/B. An overhead crane is provided to move the waste from the de-watering area to the storage area and to retrieve waste from storage for loading onto trucks for shipment offsite for disposal. It is estimated that for 30 days of operation, about three containers of Class B waste and 20 containers of Class A waste will be generated. The number of shipments is determined by their facility to support plant operations. Long-term radioactive waste storage is a site-specific requirement. The COL applicant is required to provide information to adequately store the radioactive waste.

Normally, filled waste containers are shipped promptly after they are filled. If a shipment cannot be promptly arranged, or if a single shipment is not cost-effective, the waste containers are staged in the shielded waste storage area. Waste containers can be retrieved from the storage area when a shipment is arranged. Waste containers are loaded for shipment inside the truck bay area in a controlled environment, minimizing radiation doses.

Operating procedures and administrative controls are implemented to prevent or minimize the use of listed or characteristic chemicals. If mixed waste is generated, it is collected primarily in 55-gallon drums and sent offsite to an appropriately licensed processor. When circumstances dictate the storage or disposal of mixed waste, those operations will be in accordance with the applicable regulatory requirements and associated permits.

11.4.2.4 Effluent Controls

The SWMS process flow diagrams are presented as follows:

- Figure 11.4-1 Dry active waste and spent filter handling subsystem
- Figure 11.4-2 Spent resin and charcoal handling subsystem
- Figure 11.4-3 Oil and sludge handling subsystem

Design process flow rates, tank capacities, and key instruments are included in these flow diagrams to indicate method of operation, system interfaces and bypass routes. The SRSTs are designed to store spent resin between fuel cycles (23.5 months) and the radioactive waste storage capacity is for 30 days.

11.4-17 Mobile Low-level Radioactive Waste Processing System, ~~ANSI 40.37-200~~ANSI/ANS-40.37-2009.

11.4-18 Deleted

11.4-19 Waste classification, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 61.55.

11.4-20 Waste characteristics, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 61.56.

11.4-21 Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20, Appendix G.

11.4-22 ~~Licensing requirements for land disposal of radioactive waste, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 61.~~Deleted

11.4-23 Nuclear Energy Institute, Generic FSAR Template for Process Control Program, NEI 07-10A, Revision 0.

11.4-24 Dose limits for individual members of the public, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20.1301.

11.4-25 Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, NUREG-0133.

11.4-26 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109, Rev. 1, October 1977.

11.4-27 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Regulatory Guide 1.111, Rev. 1, July 1977.

11.4-28 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Regulatory Guide 1.113, Rev. 1, April 1977.

11.4-29 ~~Deleted.~~ Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment, U.S. Nuclear Regulatory Commission, IE Bulletin No. 80-10, May 6, 1980.

11.4-30 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800.

11.4-31 Packaging and transportation of radioactive material, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 71.

Chapter 12

US-APWR DCD Chapter 12 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.1-8	12.1.3 2nd paragraph, 1st sentence	Replaced “In addition, the COL Applicant is to provide …” with “In addition, the COL Applicant is to describe …”. Editorial correction
12.1-8	12.1.4 COL 12.1(5)	Replaced “The COL Applicant is to provide …” with “The COL Applicant is to describe …”. Editorial correction
12.1-8	12.1.4 COL 12.1(6)	Replaced “The COL applicant is to perform periodic review of operational practices to ensure …” with “The COL applicant is to describe the periodic review of operational practices to ensure …”. Editorial correction
12.1-8	12.1.4 COL 12.1(7)	Replaced “The COL applicant is to track implementation of requirements for record retention according to …” with “The COL applicant is to describe how implementation of requirements for record retention are tracked according to …”. Editorial correction
12.1-8	12.1.4 COL 12.1(8)	Added the sentences “The COL applicant is responsible for the development of the operational procedures, following the guidance of RG 4.21 (Reference 12.1-27), for the operation and handling of all structure, system, and components (SSC) which could be potential sources of contamination within the plant. These procedures will be developed according to the objective of limiting leakage and the spread of contamination within the plant. Reason: Reflected the additional comment of COL applicant
12.2-7	12.2.1.1.10 8th paragraph, 1st sentence	Replaced “… are to be provided by the COL Applicant.” with “… are to be described by the COL Applicant.”. Editorial correction
12.2-20	Table 12.2-1 (Sheet 6 of 6) 1st column, 5th row	Replaced “B.A. evaporator feed demineralizer” with “B.A. evaporator feed demineralizer filter”. Editorial correction
12.2-52	Table 12.2-33 2nd column, 7th row	Replaced “1.0E+01” with “1.1E+01”. Editorial correction
12.2-108	Figure 12.2-2, title	Replaced “… Gamma Ray Energy Flux …” with “… Gamma Ray Flux …”. Editorial correction

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.3-28	12.3.4 6th paragraph, 1st sentence	Replaced "… are to be provided by the COL Applicant." with "… are to be described by the COL Applicant.". Editorial correction
12.3-29	12.3.4.1.1 1st paragraph, 3rd bulleted item	Replaced "To provide local and remote indication of ambient gamma ray …" with "To provide local and remote indication of ambient gamma radiation …". Editorial correction
12.3-29	12.3.4.1.2 2nd paragraph, 2nd bulleted item	Replaced "Areas which are normally accessible and …" with "Areas which are normally accessible or …". Editorial correction
12.3-37	12.3.6 COL 12.3(1)	Replaced "The COL Applicant is responsible for the use of portable instruments, …" with "The COL Applicant shall describe portable instruments, …". Editorial correction
12.5-1	12.5 1st paragraph, 1st sentence	Replaced "The COL Applicant is to provide …" with "The COL Applicant is to describe …". Editorial correction

In addition, the COL Applicant is to ~~provide~~describe the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. This program is to be developed, implemented and maintained as described in the Nuclear Energy Institute Technical Report, NEI 07-03A (Reference 12.1-25), including compliance with the relevant quality assurance guidance provided in RG 1.33 (Reference 12.1-26). The specific CFR criteria referenced in NEI 07-03A shall be met and strictly adhered to. All recommendations and guidance referenced in NEI 07-03A are to be addressed and implemented as applicable to the US-APWR and the plant site.

Operational procedures will be developed, following the guidance of RG 4.21 (Reference 12.1-27), for the operation and handling of all structure, system, and components (SSC) which could be potential sources of contamination within the plant. These procedures will be developed according to the objective of limiting leakage and the spread of contamination within the plant. See Subsection 12.1.4 for COL information.

12.1.4 Combined License Information

COL 12.1 (1) The COL Applicant is to demonstrate that the policy considerations regarding plant operations are compliance with RG 1.8, 8.8 and 8.10 (Subsection 12.1.1.3).

COL 12.1 (2) Deleted.

COL 12.1 (3) The COL Applicant is to describe how the plant follows the guidance of RG 8.2, 8.4, 8.6, 8.7, 8.9, 8.13, 8.15, 8.25, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36 and 8.38.

COL 12.1 (4) Deleted.

COL 12.1 (5) The COL Applicant is to ~~provide~~describe the operational radiation protection program for ensuring that occupational radiation exposures are ALARA.

COL 12.1 (6) The COL applicant is to describe the~~perform~~ periodic review of operational practices to ensure configuration management, personnel training and qualification update, and procedure adherence.

COL 12.1 (7) The COL applicant is to ~~track~~describe implementation of requirements for record retention are tracked according to 10 CFR 50.75(g) and 10 CFR 70.25(g) as applicable.

COL 12.1(8) The COL applicant is responsible for the development of the operational procedures, following the guidance of RG 4.21 (Reference 12.1-27), for the operation and handling of all structure, system, and components (SSC) which could be potential sources of contamination within the plant. These procedures will be developed according to the objective of limiting leakage and the spread of contamination within the plant.

12.2.1.1.10 Miscellaneous Sources

The principal sources of activity outside the buildings but inside the tank house include the following:

- The refueling water storage auxiliary tank
- The primary makeup water tank

The content of the water tanks is processed by the SFP purification system, or the boron recycle system until the activity in the fluids is sufficiently low to result in dose rates less than 0.25 mrem/h at 2 meters from the surface of the tank.

Radionuclide inventories of the refueling water storage auxiliary tank and primary makeup water tank are presented in Tables 12.2-50 and 12.2-51. There are no other significant amounts of radioactive fluids permanently stored outside the buildings.

Spent fuels are stored in the SFP. When the fuel is to be moved away from the SFP, it is placed in a spent fuel shipping cask for transport.

Storage space is allocated in the radwaste processing facility for storage of spent filter cartridges and packaged spent resins.

Radioactive wastes stored inside the plant structures are shielded so that areas outside the structures meet Radiation Zone I criteria. Additional storage space for radwaste is to be provided in the detailed design by the COL Applicant. If it becomes necessary to temporarily store radioactive wastes/materials outside the plant structures, radiation protection measures are to be taken by the radiation protection staff to ensure compliance with 10 CFR 20 (Reference 12.2-1), 40 CFR 190 (Reference 12.2-6) and to be consistent with the recommendations of RG 8.8 (Reference 12.2-2).

The SWMS facilities process and store dry active waste. If it becomes necessary to install additional radwaste facilities for dry active waste, it is to be provided by the COL Applicant. Radiation shielding is to be provided such that the dose rates comply with the requirements of 10 CFR 20 (Reference 12.2-1) and 40 CFR 190 (Reference 12.2-6). Interior concrete shielding is provided to limit exposure to personnel during waste processing. The ALARA methodology of RGs 8.8 (Reference 12.2-2) and 8.10 (Reference 12.2-3) has been used in the design of this facility.

Any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are to be ~~provided~~described by the COL Applicant.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent fuel, the residual heat removal system (RHRS), and the incore instrumentation system (ICIS). Individual components may

Table 12.2-1 Radiation Sources Parameters (Sheet 6 of 6)

Components	Assumed Shielding Sources									
	Source Approximate Geometry as Annular Cylinder Volume			Source Characteristics					Quantity
	Outer Radius (in.)	Inner Radius (in.)	Height (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	Designed Upper limit dose rate (mrem/h)	
Auxiliary Building									
Reactor coolant Filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	2
Mixed bed demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	3
B.A. evaporator feed demineralizer filter	3.4	2.7	19.7	Homogeneous	Water	62.4	Ignored	100	1
Boric acid filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	1
Seal water injection filter	1.7	1.6	19.9	Homogeneous	Water	62.4	Ignored	100	2
Waste effluent inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SFP filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SG blowdown demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	2

Table 12.2-33 Spent Fuel Pit Demineralizer Sources (70 ft^3 of Resin)

Spent Fuel Pit Demineralizer activity	
Nuclide	Activity (μCi/cm^3)
Co-60	1.3E+01
Spent Fuel Pit Demineralizer source strength	
Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	8.0E-01
0.3	1.1~~1.0~~E+01
0.8	3.0E+01
1.0	4.9E+05
1.5	7.3E+05
2.0	1.1E+01
3.0	5.2E-02

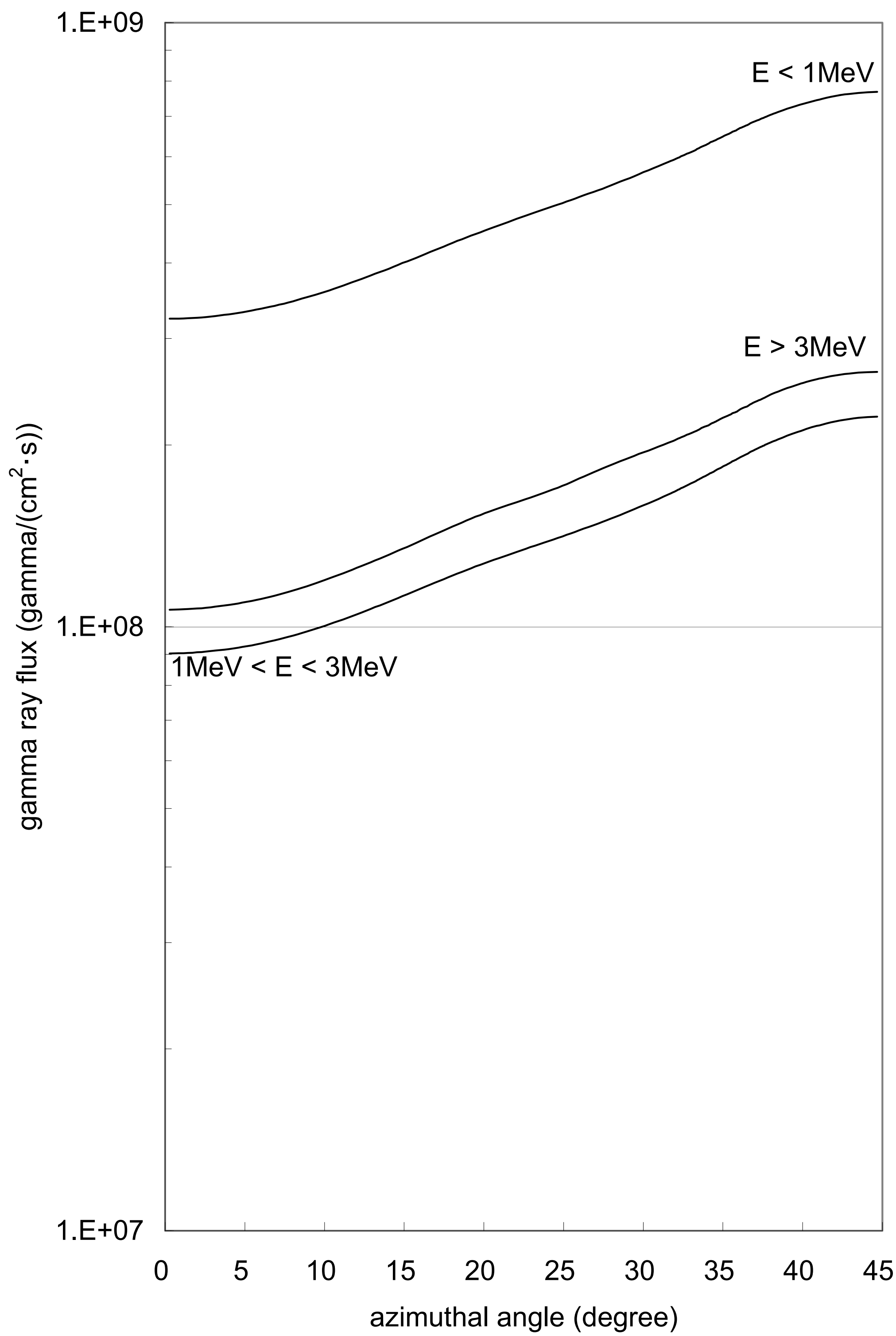


Figure 12.2-2 Azimuthal Distribution of Gamma Ray ~~Energy~~ Flux Incident on the Primary Shield at the Reactor Core Midplane

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The radiation monitoring system consists of the following:

- Area Radiation Monitoring System (ARMS)
- Airborne Radioactivity Monitoring System
- Process and Effluent Radiation Monitoring System
- Sampling system
- Post-Accident Monitoring Systems (PAM) radiation monitors

The process and effluent radiation monitoring system and sampling systems are described in Chapter 11, Section 11.5.

The PAM are described in Chapter 7, Section 7.5. The portable dose rate and activity monitoring instruments are Type E PAM.

The ARMS and Airborne Radioactivity Monitoring System supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and assure compliance with the personnel radiation protection requirements of 10 CFR 20 (Reference 12.3-2), 10 CFR 50 (Reference 12.3-7), 10 CFR 70 (Reference 12.3-21), and the guidelines of RGs 1.21 (Reference 12.3-22), 1.97 (Reference 12.3-23), 8.2 (Reference 12.3-24), and 8.8 (Reference 12.3-1), ANSI N13.1-1999 (Reference 12.3-25), and IEEE 497-2002 (Reference 12.3-28).

The design of the fuel pool racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 50.68 (Reference 12.3-26), are not needed.

The ARMS are in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 12.3-27).

The use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737, are to be ~~provided~~described by the COL Applicant.

12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design Objectives

The design objectives of the ARMS during normal operating plant conditions and anticipated operational occurrences are as follows:

- To record radiation levels in specific areas of the plant

- To warn of uncontrolled or inadvertent movement of radioactive material in the plant
- To provide local and remote indication of ambient gamma ~~ray~~radiation and local and remote alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area
- To furnish information for making radiation surveys

By meeting the above objectives, the ARMS aids health physics personnel in keeping radiation exposures ALARA.

The design objectives of the ARMS during postulated accidents are as follows:

- To provide the capability to alarm and initiate a containment ventilation isolation signal in the event of a LOCA or abnormally high radiation inside the containment (monitors RMS-RE-091, RMS-RE-092, RMS-RE-093, and RMS-RE-094). In Modes 1 through 4, four trains of radiation monitors are required to ensure radiation-monitoring instrumentation necessary to initiate the containment ventilation isolation.
- To provide long-term post-accident monitoring (Chapter 7, Section 7.5)

12.3.4.1.2 Criteria for Location of Area Radiation Monitors

The locations of the area radiation monitors are shown in Figure 12.3-1.

Considerations for area radiation monitor locations include:

- Areas which are normally accessible, and where changes in plant conditions can cause significant increases in personnel exposure rate above that expected for the area
- Areas which are normally accessible ~~and~~or occasionally accessible where a significant increase in exposure rate resulting from operational transients or maintenance activities may occur
- The containment area where the level of radioactivity needs to be monitored to detect the presence of fission products during a DBA
- Area monitor detectors are located such that inadvertent shielding by structural materials is minimized
- In the selection of area monitors, consideration is given to the range of temperature, pressure and humidity of the areas where the detectors or electronics are located

The ARMS provides a continuous, direct indication or recording of radiation levels in the control room and raises alarms locally and in the control room when radiation levels exceed the set values.

12.3.6 Combined License Information

COL 12.3(1) The COL Applicant ~~is responsible for the use of~~shall describe portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.

COL 12.3(2) Deleted.

COL 12.3(3) Deleted.

COL 12.3(4) The COL Applicant is to provide the site radiation zones that is shown on the site-specific plant arrangement plan.

COL 12.3(5) The COL Applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.

COL 12.3(6) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69.

COL 12.3(7) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.

COL 12.3(8) IF the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.

COL 12.3(9) In order to ensure that the B.A. evaporator room does not become a VHRA during the end of cycle, the COL Applicant is to stipulate a need for routine surveillance in the Radiation Protection Program. In the event that the routine surveillance shows an increase in dose level, the COL Applicant must provide an appropriate strategy to sufficiently reduce the dose rate below the criteria for a VHRA.

COL 12.3(10) The COL Applicant will address the site-specific design features, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21.

12.5 Operational Radiation Protection Program

The COL Applicant is to ~~provide~~describe the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. This Combined Information is addressed in Subsection 12.1.4. This program will be based on RG 1.206 and any additional guidance developed by the industry and approved by the NRC.

The program consists of the following:

- A detailed management policy
- An organizational structure with clearly defined responsibilities
- Definition and description of all facilities, including laboratories and office spaces
- Definition and description of the monitoring instrumentation and equipment(note)
- Definition and description of the personnel protective clothing and equipment, including the necessary inventory of supplies
- Definition and description of other protective equipment, such as portable ventilation systems, temporary shielding, etc.
- Procedures on radiological surveillance
- Procedures on methods to maintain exposures ALARA
- Procedures on posting and labeling
- Procedures on access control
- Procedures on radiation work permits
- Procedures on personnel monitoring
- Procedures on dose control
- Procedures on contamination control
- Procedures on respiratory protection
- Procedures on radioactive material control
- Procedures on radiation protection training
- Quality assurance programs in effect

Note: This includes the personnel monitoring and radiation survey equipment, and laboratory equipment used to analyzed or measure radiation levels and radioactivity concentrations.

Chapter 13

US-APWR DCD Chapter 13 Rev. 2, Tracking Report Rev. 7 Change List

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
13.3-1	Section 13.3 TSC 1^{st} sentence	Editorial correction: Delete “onsite.”
13.3-1	Section 13.3 TSC 3^{rd} bullet	Editorial correction: Change: “The facility consists of a plant data display system by visual display units and large display panel. These equipments are redundant including these power supplies.” to “The facility provides a plant data display system consisting of visual display units and large display panel. These equipments (i.e. visual display units and large display panel) are redundant including their power supplies.
13.3-1	Section 13.3 TSC 4^{th} bullet	Editorial correction: Change: “The TSC provides telephones and facsimiles, which utilize by multiple methods of telecommunication including private and public lines, satellite communications, and ample working areas for all personnel as described in section 9.5.2.” to “The TSC provides telephones and facsimile machines, including land-line, cellular, and satellite communication capabilities, which utilize by multiple methods of telecommunication including private and public lines, satellite communications. Ample working areas for all personnel as described in section 9.5.2.
13.3-2	Section 13.3 EOF 1^{st} sentence	Editorial correction: Delete “near site or on site.”
13.3-3	Section 13.3 Data communication with the TSC, the EOF, and the ERDS Last sentence before SPDS	Editorial correction: Change “evolves” to “evolve.”
13.3-4	Section 13.3 4^{th} line	Editorial correction: Change “planning” to “response.”

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
13.3-4	Section 13.3.1 1st paragraph	Editorial correction: Change: "The plan is a physically separate document (see section 13.3 of the Final Safety Analysis Report (FSAR)) and incorporates, by reference, State and local emergency plans." to "The plan is a physically separate document (See Section 13.3 of the Final Safety Analysis Report (FSAR)) and developed in accordance with 10 CFR 50.47, Appendix E and 10 CFR 52."
13.3-4	Section 13.3-1 2nd paragraph	Editorial correction: Change "FSAR" to "emergency plan of COL Application."
13.6-8	Reference 13.6-8	Delete the Technical Report MUAP-08003, Cyber Security Program, which was withdrawn by the NRC.

13.3 Emergency Planning

Emergency planning is designated as the responsibility of the COL Applicant. However, design features, facilities, functions, and equipment necessary for emergency planning are considered in the design bases for the standard plant (Ref. 13.3-1, 13.3-2, 13.3-3, 13.3-4, 13.3-5). Details of these features, as they relate to the basic design, are included as follows and are consistent with the descriptions in Chapter 7. Interfaces of these features with site-specific designs and site parameters are the responsibility of the COL Applicant.

- Technical Support Center (TSC)

  The ~~onsite~~ TSC is an onsite facility that provides plant management and technical support to the plant operations personnel during emergency conditions as described in subsection 7.5.1.6.1. The TSC has technical data displays and plant records available to assist in the detailed analysis and diagnosis of abnormal plant conditions and any significant release of radioactivity to the environment. The TSC provides the following functions:

  - The TSC has facilities to support the plant management and technical personnel who are assigned there during an emergency.

  - The TSC is the primary onsite communications center for the plant during an emergency.

  - The facility ~~consists of~~provides a plant data display system ~~by~~consisting of visual display units and large display panel. These equipments (i.e. visual display units and large display panel) are redundant including ~~these~~their power supplies. The TSC displays include,

    - Plant systems variables,
    - In-plant radiological information,
    - Meteorological information, and
    - Offsite radiological information.

  - The TSC provides telephones and facsimile~~s~~ machines, including land-line, cellular, and satellite communication capabilities, which utilize by multiple methods of telecommunication including private and public lines, satellite communications. ~~, and a~~Ample working areas for all personnel as described in section 9.5.2.

  - The TSC is close to the main control room (MCR), located in the access building (AC/B). The walking time from the TSC to the MCR does not exceed 2 minutes.

  - Working space, without crowding, for the personnel assigned to the TSC at the maximum level of occupancy is approximately 75 square feet per person.

- ° The TSC working space is sized for a minimum of 25 persons including 20 persons designated by the licensee and five NRC personnel. The size and layout of TSC gives necessary space to maintain and repair TSC equipment, and is sufficient for storage of plant records and historical data.

- – A TSC heating, ventilation, and air conditioning (HVAC) system is provided that includes high-efficiency particulate air and charcoal filters as described in subsection 9.4.3.1.2.4.

  - ° The TSC HVAC system functions in a manner comparable to the main control room HVAC system.

  - ° The TSC HVAC need not be seismic Category I qualified, redundant, instrumented in the MCR, or automatically activated to fulfill its role.

- Emergency Operations Facility (EOF)

  The EOF is a ~~near-site or on-site~~ support facility for the management of overall licensee emergency response (including coordination with Federal, State, and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective actions.

  - – The EOF has the appropriate technical data displays and plant records to assist in the diagnosis of plant conditions and to evaluate the potential or actual release of radioactive materials to the environment.

  - – The EOF computer provides plant data displays to assist in the diagnosis of plant conditions and to evaluate the potential or actual release of radioactive materials to the environment.

  - – A senior licensee official in the EOF organizes and manages licensee offsite resources to support the TSC and the MCR operators.

- Emergency Response Data System (ERDS)

  The ERDS is a data transmission system. This system is designed to send a set of variables from the plant to the NRC Operations Center. This data may be used for analyses by the NRC headquarters technical support groups and the NRC executive team. The ERDS provides for the following functions:

  - – This system fulfills the function of the emergency response data system of Appendix E to Title 10, Code of Federal Regulations (CFR), Part 50 (Ref. 13.3-1).

  - – This system transmits information that aids the NRC in its role of providing advice and support to the nuclear power plant licensee, State and local authorities, and other Federal officials.

- Data communication with the TSC, the EOF, and the ERDS

- A data communication system establishes the interface and link with the TSC, the EOF, and the ERDS and allows data exchange with the plant. The TSC receives plant information from the unit bus.

- The EOF and the ERDS receive plant information from the station bus.

- The following countermeasures are applied to prevent cyber security threats:

  - The plant instrumentation and control (I&C) and HSI systems do not link to external networks. An exception is the link from unit management computer to the station bus.

  - Communication from the unit management computer to the station bus is restricted one direction. A dedicated transmission protocol is used which is not general-purpose, such as transmission control protocol/internet protocol, user datagram protocol, etc.

  - Communication between the station bus and the TSC, the EOF or the ERDS (NRC) is also one direction and uses a dedicated transmission protocol.

  - If a computer system, which has a general-purpose local area network, is connected to the station bus, an adequate gateway processor with a firewall function is inserted.

  - The firewall program currently used is MISTY®, which uses 128-bit code key. This firewall program is safer than the data encryption standard code, which is more typically used in the U.S. Alternate firewall programs may be used in the future, as the security features of new technology evolve~~s~~.

- Safety Parameter Display System (SPDS)

  The SPDS provides a display of plant parameters from which the safety status of operation may be assessed in the MCR, the TSC, and the EOF. The SPDS provides the following functions:

  - The primary function of the SPDS is to help operating personnel in the MCR make quick assessments of the plant safety status.

  - Duplication of the SPDS displays in the TSC and the EOF improves the exchange of information between these facilities and the MCR and assists corporate and plant management in the decision-making process.

  - The SPDS is operated during normal operations and during all classes of emergencies.

  - The SPDS has the flexibility to allow future modifications to be incorporated, such as the capability to handle operator interaction and diagnostic analysis.

- The functions and design of SPDS in the MCR are realized as a part of the overall HSI design

The Postaccident Sampling System (PASS) is provided for emergency ~~planning~~response. It is described in section 9.3.2 and 12.3.

- Other Emergency Facilities

Personnel and equipment decontamination facilities for normal operation are located in the Access Building and described in Subsection 12.3.1.1.2. These facilities would be used in emergency conditions as part of the site Emergency Plan.

13.3.1 Combined License Application and Emergency Plan Content

The development of a comprehensive emergency plan shall be designated as the responsibility of the COL Applicant. The plan is a physically separate document (See ~~S~~section 13.3 of the Final Safety Analysis Report (FSAR)) and ~~incorporates, by reference, State and local emergency plans~~developed in accordance with 10 CFR 50.47, Appendix E and 10 CFR 52. It includes copies of letters of agreement from State and local governmental agencies with emergency planning responsibilities.

The ~~FSAR~~emergency plan of COL Application addresses emergency classification and action level scheme. It also addresses security-related aspects of emergency planning.

13.3.2 Emergency Plan Considerations for Multi-Unit Site

The development of the emergency plan for multi-unit site is designated as the responsibility of the COL Applicant depending on the location of the new reactor on, or near, an operating reactor site with an existing emergency plan.

13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria

The development of emergency planning inspections, tests, analyses, and acceptance criteria are designated as the responsibility of the COL Applicant.

13.3.4 Combined License Information

COL 13.3(1) *The COL Applicant is to develop interfaces of design features with site specific designs and site parameters.*

COL 13.3(2) *The COL Applicant is to develop a comprehensive emergency plan as a physically separate document.*

COL 13.3(3) *The COL Applicant is to develop an emergency classification and action level scheme.*

13.6-5 'Requirements for the Protection of Safeguards Information,' "Physical Protection of Plants and Materials," Energy. Title 10, Code of Federal Regulations, Part 73.21, U.S. Nuclear Regulatory Commission, Washington, DC.

13.6-6 US-APWR Design Certification Physical Security Element Review (Safeguards Information), Rev. ~~2~~3, ~~November~~October ~~2009~~2010

13.6-7 US-APWR High Assurance Evaluation Assessment (Safeguards Information), Rev. ~~1~~2, ~~November~~October ~~2009~~2010

~~13.6-8 US-APWR Cyber Security Program, MUAP-08003-P, Rev. 1, August 2008~~

Chapter 14

US-APWR DCD Chapter 14 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
14-i	Contents 14.2.1.2	ITP was replaced with Test Program
14.2-2	14.2.1	“construction,” was deleted Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
14.2-3	14.2.1.2	ITP was replaced with Test Program
14.2-77 to 78	14.2.12.1.46	Added “5. A report exists that demonstrates the reliability of the alternate ac power sources meets or exceeds 95% as determined in accordance with NSAC-108 (Reference 8.4-2) or equivalent methodology to meet the Criterion 5 of Section C.3.3.5, RG 1.155, based on historical data of the similar type of the ac alternate power sources.” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
14.3-59 trough 64	Table 14.3-2 (sheet 1 through 4 of 4)	Revised based on the ITAAC improvement activity with the COL applicant.

CONTENTS

- Assumed to function or which are credited in the accident analysis of the US-APWR DCD or applicable Final Safety Analysis Report (FSAR)
- Required to process, store, control, and/or limit the release of radioactive materials
- Used in the special low-power testing program to be conducted at power levels no greater than five percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program as required for resolution of Three Mile Island (TMI) action plan item I.G.1 (Reference 14.2-7)
- Identified as risk-significant as discussed and identified in Subsection 17.4.7 and Table 17.4-1

The ITAAC required by 10CFR 52.47(b)(1) (Reference 14.2-8) for the US-APWR design are found in the Tier 1 document. The criteria for ITAAC selection are contained in Section 14.3, inspections, tests, analyses, and acceptance criteria.

The ITP consists of ~~construction,~~ preoperational~~,~~ and startup tests.

Following the plant construction, testing is accomplished to demonstrate the proper performance of SSCs and design features.

Preoperational tests do not begin until construction and designated construction tests of the system are essentially completed. Preoperational tests are performed in cold conditions and at elevated temperatures produced by reactor coolant pump and pressurizer heater operation.

The initial fuel loading marks the beginning of startup testing. Startup tests as defined by Subsection 14.2.1.2.3 are performed to demonstrate that plant systems meet the performance requirements and that the plant can operate in an integrated fashion.

The preparation and performance of preoperational and startup tests are the responsibility of the COL licensee.

The ITP described in this chapter only addresses those systems and components within the US-APWR. A description of the program for the testing of other components and systems that are site-specific is discussed in US-APWR Test Program Description Technical Report, MUAP-08009 (Reference 14.2-29). Testing of these items demonstrates that they meet requirements as defined in the Final Safety Analysis Report (FSAR).

14.2.1.1 Test Program for Nuclear and Balance of Plant Systems

Preoperational and startup testing is conducted in accordance with an approved manual containing ITP administrative controls. The manual is prepared by the startup organization. Final approval of the manual is by the designated plant management. The preparation and approval of the manual is completed prior to the preparation of the first test procedure. This manual contains the administrative procedures and requirements that govern the activities of the startup organization and its interface with other organizations.

The procedures within the manual perform the following functions:

- Provide the organization of the startup organization and staffing (Subsections 14.2.2 and 14.2.2.1).
- Describe the preoperational and startup test procedure preparation, review, and approval (Subsection 14.2.3).
- Describe the conduct of testing of the ITP and the controls (Subsection 14.2.4)
- Specify the process for the evaluation, review and approval of the individual test results (Subsection 14.2.5).
- Specify the retention period of the test results and describe how the ITP results are compiled and maintained (Subsection 14.2.6).
- Establish the requirements for transitioning between test phases and between power test plateaus (Subsections 14.2.5.1, 14.2.10.1, 14.2.10.2, and 14.2.10.3).

The ITP includes tests on systems in both the nuclear portion of the plant, the balance of plant, or non-nuclear areas. The tests conducted on safety-related systems demonstrate the capability of the SSCs to meet performance requirements and design criteria. The tests on non safety-related systems verify the operability of the systems and/or components and their capability to support safety-related systems, where applicable. The testing continues through the initial fuel loading, startup, and power ascension.

Tests are performed to demonstrate the operation of each system independently and the operation of the systems in an integrated plant environment.

14.2.1.2 Major Phases of ~~ITP~~Test Program

14.2.1.2.1 Construction Tests

Construction and preliminary tests and inspections typically consist of activities such as hydrostatic pressure tests, flushing, cleaning, wiring continuity and separation checks, electrical distribution protection relays, initial instrument calibrations, valve functional checks, motor rotational checks, etc., and functional tests of components.

The objective of the construction and preliminary tests and inspections test phase is to verify and document that construction and installation of equipment in the facility have been accomplished in accordance with design, and that the equipment and components are functional and ready for preoperational testing.

Construction test abstracts are not included in this section. The development of construction and installation tests is based on engineering design, applicable industry standards and vendor information. A construction test matrix is developed for each system listing required tests and inspections for piping, wiring, equipment, valves,

3. In offsite source available condition, load sequence is tested by initiating an ECCS actuation signal.

4. Each electrical division is operated independently of other divisions and division separation is verified in accordance with RG 1.41.

5. Verify all associated indications and alarms during test sequences.

D. Acceptance Criteria

1. The PSMS, the bus undervoltage relays, and the degraded voltage relays operate in accordance with design (see Subsection 8.3.1.1.3).

2. The loading intervals for supplying from Class 1E gas turbine generator are within the design limits.

3. Each train loads are sequenced on the bus by initiating of an ECCS actuation signal.

4. Each electrical division operates independently of other divisions.

5. All associated indications and alarms operate per design.

14.2.12.1.46 Alternate ac Power Sources for Station Black Out Preoperational Test

A. Objectives

1. Demonstrate the operability of each alternate ac power source breaker and associated interlocks.

2. Demonstrate the operation of air start and fuel systems.

3. Demonstrate the ability of the alternate ac power source to synchronize with the offsite power system.

4. Determine the fuel oil consumption of each alternate ac power source while operating under continuous rating load conditions.

5. Verify that, with the alternate ac power source operating in the test mode connected to its bus, an automatic start signal overrides the test mode by returning the alternate ac power source to standby operation.

B. Prerequisites

1. Required construction acceptance tests are completed.

2. Required electrical power supplies and control circuits are operational.

3. The alternate ac power source fuel oil system is available.

4. Adequate ventilation for the alternate ac power source area is available.

5. A report exists that demonstrates the reliability of the alternate ac power sources meets or exceeds 95% as determined in accordance with NSAC-108 (Reference 8.4-2) or equivalent methodology to meet the Criterion 5 of Section C.3.3.5, RG 1.155, based on historical data of the similar type of the ac alternate power sources.

C. Test Method

1. Fuel oil is transferred from the fuel oil storage tank to the fuel oil day tanks by means of the transfer pumps. Appropriate flow parameters are recorded.

2. The control logic of the alternate ac power source breaker, alternate ac power source start circuit, and support pumps and valves are verified.

3. The operability of the alternate ac power source starter is verified.

4. The alternate ac power source is started, voltage and frequency control demonstrated, phase rotation verified, and the backup generator synchronized to offsite power and loads.

5. During the testing, fuel oil consumption is monitored with the alternate ac power source operating at the continuous load rating.

6. With a simulated LOOP signal, the proper alternate ac power source trips is verified.

7. With the alternate ac power source connected to its bus, an automatic start signal causes it to return to standby operation.

8. Verify all associated indications and alarms during test sequences.

D. Acceptance Criteria

1. The controls, interlocks, and operation of the alternate ac power source breakers and support systems operate as designed (see Subsection 8.3.1.1.1).

2. Each alternate ac power source can be synchronized with offsite power.

3. Upon the receipt of automatic start signals, the alternate ac power sources operate as designed.

4. The alternate ac power source fuel oil consumption does not exceed the design requirements.

5. All associated indications and alarms operate per design.

14.2.12.1.47 125 V dc Class 1E Preoperational Test

A. Objectives

Table 14.3-2 Example of ITAAC Table

(Sheet 1of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.a The functional arrangement of the _______system is as described in the Design Description of Subsection ______~~Design Description~~ and in Table ____and as shown on Figure _____ .	1.a ~~An i~~Inspection of the as-built _______system will be performed.	1.a The as-built ____system conforms to the functional arrangement as described in the Design Description of ~~this~~ Subsection and in Table ______and as shown in Figure ____.
1.b Each mechanical division of the _____ system (Divisions A, B, C & D) is physically separated from the other divisions so as not to preclude accomplishment of the safety function.	1.b Inspections and analysis of the as-built system will be performed.	1.b A report exists and concludes that ~~E~~each mechanical division of the _____system is physically separated from other mechanical divisions of the system by ~~structural~~spatial separation, ~~and/or fire~~ barriers, or enclosures so as to assure that the functions of the safety related system are maintained.
2a. The ASME Code Section III components of the___system identified in Table _____are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.	2.a. ~~An i~~Inspection of the as-built ASME Code Section III components of the _____ system, identified in Table ___ will be performed.	2.a. The ASME Code Section III data report(s) (certified, when required by ASME Code) and inspection reports (including N-5 Data Reports where applicable) exist and conclude that the as-built ASME Code Section III components of the ____system identified in Table _____ are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.b The ASME Code Section III components of the _____ system identified in Table ______ are reconciled with the design requirements.	2.b A reconciliation analysis of the components identified in Table _____using as-designed and as-built information and ASME Code Section III design report(s) (NCA-3550) will be performed.	2.b The ASME Code Section III design report(s) (certified, when required by ASME Code) exist and concluded that the design reconciliation has been completed in accordance with the~~as-built~~ ASME Code Section III for the as-built components of the ____system identified in Table ____~~are reconciled with the design requirements~~. The report documents the results of the reconciliation analysis.

Table 14.3-2 Example of ITAAC Table

(Sheet 2of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3.a Pressure boundary welds in ASME Code Section III components, identified in Table _____, meet ASME Code Section III requirements for non-destructive examination of welds.	3.a Inspections of the as-built pressure boundary welds in ASME Code Section III components, identified in Table ____, will be performed in accordance with the ASME Code Section III.	3.a The ASME Code Section III code reports exist and conclude that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds in ASME Code Section III components identified in Table ______.
4.a The ASME Code Section III components, identified in Table _____, retain their pressure boundary integrity at their design pressure.	4.a A hydrostatic test will be performed on the components, identified in Table ______, required by the ASME Code Section III to be hydrostatically tested.	4.a ASME Code Data Report(s) exist and conclude that ~~T~~the results of the hydrostatic test of the components identified in Table _____ as ASME Code Section III conform with the requirements of the ASME Code Section III.
5.a The seismic Category I equipment, identified in Table , ~~is designed to~~ can withstand seismic design basis loads without loss of safety function.	5.a.i Inspections will be performed to verify that the seismic Category I equipment identified in Table _____is located in ~~the containment vessel and reactor building~~a seismic Category I structure.	5.a.i The seismic Category I equipment identified in Table _____is located in ~~the containment vessel and reactor building~~ a seismic Category I structure.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.a.ii Type tests, analyses, or a combination of type tests and~~/or~~ analyses of seismic Category I equipment identified in Table ____ will be performed using analytical assumptions, or will be performed under conditions, which bound the seismic design basis requirements.	5.a.ii A report exists and~~The results of the type tests and/or analyses~~ concludes that the seismic Category I equipment identified in Table ____ can withstand seismic design basis loads without loss of safety function.
	5.a.iii Inspections and analyses will be performed to verify that~~on~~ the as-~~installed~~built seismic Category I equipment identified in Table ____ including anchorages, is seismically bounded by the tested or analyzed conditions.	5.a.iii A report exists and concludes that ~~T~~the as-built seismic Category I equipment identified in Table ____, including anchorage, is seismically bounded by the tested or analyzed conditions.

Table 14.3-2 Example of ITAAC Table

(Sheet 3of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b ~~Each of t~~The seismic category piping, including supports, identified in Table ~~is designed to~~can withstand ~~combined normal and~~ seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table ___ ~~are~~is supported by a seismic Category I structure(s).	5.b.i ~~Reports(s) document that each of t~~The as-built seismic Category I piping, including supports, identified in Table____ is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table ____ can withstand ~~combined normal and~~ seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that ~~each of~~ the as-built seismic Category I piping, including supports, identified in Table____ can withstand ~~combined normal and~~ seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table ____ as being qualified for a harsh environment ~~is designed~~can to withstand the environmental conditions that would exist before, during, and following a design basis accident~~event~~ without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses using the design environmental conditions, or under conditions which bound the design environmental conditions, will be performed on Class 1E equipment identified in Table ____ as being qualified for~~located~~ in a harsh environment.	6.a.i A report exists and~~The results of the type tests, and/or analyses, or a combination of type tests and analyses~~ concludes that the Class 1E equipment identified in Table ____as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident~~event~~ without loss of safety function for the time required to perform the safety function.

Table 14.3-2 Example of ITAAC Table

(Sheet 4of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	6.a.ii Inspection will be performed on the as-built Class 1E equipment identified in Table ____ as being qualified for a harsh environment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table ____as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table ______, is powered from ~~their~~its respective Class 1E division.	6.b A ~~T~~tests~~s~~ will be performed on each division of the as-built Class 1E equipment identified in ~~the~~Table ~~system~~ by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the Class 1E equipment identified in Table ____under test.
6.c Separation is provided between redundant divisions of ________ system Class 1E ~~divisions~~cables, and between Class 1E ~~divisions~~cables and non-Class 1E cables.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided in accordance with RG 1.75, between the as-built cables of redundant ________ system Class 1E divisions and between Class 1E cables~~divisions~~ and non-Class 1E cables.

Chapter 15

US-APWR DCD Chapter 15 rev. 2, Tracking Report Rev. 7 Change List

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.0-24	Reference 15.0-13 Reference 15.0-14	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.1-123	Reference 15.1-2	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.2-81	Reference 15.2-4	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.3-55	Reference 15.3-1	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.4-74	Last bullet in Subsection 15.4.8.5.1	Editorial Added period at end.
15.4-79	Second row in table 15.4.8-3 (Sheet 1 of 3)	Editorial Replaced "%" with "MWt" to correct unit.
15.4-96	Reference 15.4-2 Reference 15.4-6	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.5-12	Reference 15.5-2	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.6-15	First sentence of second paragraph in Subsection 15.6.2.5	Editorial Deleted "with" after "operating".
15.6-17	Sixth bullet in Subsection 15.6.2.5.2	Editorial Added space before "(HVAC)".
15.6-17	First sentence of third paragraph in Subsection 15.6.2.5.3	Editorial Replaced "subsection" with "Section".

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-18	Fourth row in Table 15.6.2-1	Editorial Corrected subscript.
15.6-69	Forth item of second paragraph in subsection 15.6.5.3.1.2 Small Break LOCA Evaluation Model	Technical Reflection of the modification to the M-RELAP5 code for the small break LOCA analysis Change the description for the choked flow calculation.
15.6-74	First and second item of first paragraph of this page in subsection 15.6.5.3.2.2 Small Break LOCA	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change the word “upside” to “downside”.
15.6-78	Second paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change the word “81” to “78”.
15.6-78	Second paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change the word “122” to “124”.
15.6-78	Third paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change the word “142” to “141”.
15.6-79	Forth paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change the word “773” to “761”.

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-79	Forth paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "136" to "137".
15.6-79	Item 1. of sixth paragraph in subsection 15.6.5.3.3.2 Results of 7.5-inch Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "773" to "761".
15.6-80	Fifth paragraph in subsection 15.6.5.3.3.2 Results of 1-ft2 Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "90" to "89".
15.6-80	Fifth paragraph in subsection 15.6.5.3.3.2 Results of 1-ft2 Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "1323" to "1302".
15.6-80	Fifth paragraph in subsection 15.6.5.3.3.2 Results of 1-ft2 Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "169" to "161".
15.6-80	Seventh paragraph in subsection 15.6.5.3.3.2 Results of 1-ft2 Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change "upside" to "downside".

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-80	Item 1. of seventh paragraph in subsection 15.6.5.3.3.2 Results of 1-ft2 Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “1323” to “1302”.
15.6-81	Third paragraph in subsection 15.6.5.3.3.2 Results of DVI-Line Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Delete “Downward flow does not occur in this particular case. Upward flow through the core is maintained. The core flow is sufficient to prevent any uncovery of the core.”
15.6-81	Fifth paragraph in subsection 15.6.5.3.3.2 Results of DVI-Line Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “This figure shows that the PCT does not occur in the DVI line break, indicating that the core keeps covered throughout the transient.” to “This figure shows that the PCT of 789°F occurs at 1505 seconds in the DVI-line break, indicating that the core keeps covered throughout the transient. The PCT is significantly lower than 2200°F.”
15.6-81	Fifth paragraph in subsection 15.6.5.3.3.2 Results of DVI-Line Small Break LOCA Analysis	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “For the DVI line break, no heatup occurs. This obviously demonstrates that the regulatory limit has been met.” to “The PCT of 789°F presented in Table 15.6.5-14 indicates that this regulatory limit has been met.”
15.6-101	Table 15.6.5-9 6^{th} Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “11.9” to “11.8”.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-101	Table 15.6.5-9 9th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “84” to “78”.
15.6-101	Table 15.6.5-9 11th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “122” to “124”.
15.6-101	Table 15.6.5-9 13th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “136” to “137”.
15.6-101	Table 15.6.5-9 14th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “142” to “141”.
15.6-101	Table 15.6.5-9 16th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “299” to “317”.
15.6-102	Table 15.6.5-10 2nd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “773” to “761”.

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-103	Table 15.6.5-11 10th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “90” to “89”.
15.6-103	Table 15.6.5-11 15th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “169” to “161”.
15.6-103	Table 15.6.5-11 16th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “339” to “356”.
15.6-104	Table 15.6.5-12 2nd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “1323” to “1302”.
15.6-105	Table 15.6.5-13 3rd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “25.9” to “25.8”.
15.6-105	Table 15.6.5-13 4th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “27.7” to “27.6”.

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-105	Table 15.6.5-13 5th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “27.7” to “27.6”.
15.6-105	Table 15.6.5-13 6th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “28.9” to “28.8”.
15.6-105	Table 15.6.5-13 7th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “33.9” to “33.8”.
15.6-105	Table 15.6.5-13 13th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “not occur” to “1256”.
15.6-105	Table 15.6.5-13 14th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “lower than the initial value” to “1505”.
15.6-105	Table 15.6.5-13 15th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “N/A” to “1856”.

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-106	Table 15.6.5-14 2nd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “lower than the initial value” to “789”.
15.6-107	Table 15.6.5-15 2nd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the following: - Modification to the M-RELAP5 code for the small break LOCA analysis - Addition of the bounding bias to accumulator injection flow rate due to the scaling uncertainty Change “1174” to “1302”.
15.6-107	Table 15.6.5-15 3rd Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “1154” to “1250”.
15.6-107	Table 15.6.5-15 4th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “938” to “1220”.
15.6-107	Table 15.6.5-15 8th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “lower than the initial temperature” to “701°F”.
15.6-107	Table 15.6.5-15 10th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “756” to “715”.

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-107	Table 15.6.5-15 11th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “lower than the initial temperature” to “691°F”.
15.6-107	Table 15.6.5-15 12th Row	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Change “lower than the initial temperature” to “708°F”.
15.6-122 to 15.6-130	Figure 15.6.5-14 to 15.6.5-22	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Update Figures 15.6.5-14 to 15.6.5-22 according to re-analysis results for the 7.5-in break.
15.6-131 to 15.6-139	Figure 15.6.5-23 to 15.6.5-31	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Update Figures 15.6.5-23 to 15.6.5-31 according to re-analysis results for the 1-ft^2 break.
15.6-140 to 15.6-148	Figure 15.6.5-32 to 15.6.5-40	Technical Reflection of the re-analysis for the PCT, LMO and CWO due to the modification to the M-RELAP5 code for the small break LOCA analysis Update Figures 15.6.5-32 to 15.6.5-40 according to re-analysis results for the DVI-line break.
15.6-153	Reference 15.6-1	Editorial Incorporation of the latest revision number of the Topical Report reference.
15.6-154	Reference 15.6-14	Editorial Incorporation of the latest revision number of the Topical Report reference.

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
15.6-154	Reference 15.6-16	Editorial Incorporation of the latest revision number of the Technical Report reference.
15A-3	First sentence of first paragraph in Subsection 15A.1.2.2	Editorial Corrected subscript.
15A-10	Bottom row in table 15A-3(Sheet 1 of 2)	Editorial Deleted “.” (period) after “operating”.

15.0-5 RSIC Computer Code Collection CCC-371, ORIGEN 2.2 Isotope Generation and Depletion Code - Matrix Exponential Method, June, 2002.

15.0-6 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, 39 FR 1002, Jan. 4, 1974, as amended at 53 FR 36004, Sept. 16, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39299, July 29, 1996; 62 FR 59726, Nov. 3, 1997.

15.0-7 ECCS Evaluation Models, 10CFR 50, Appendix K.

15.0-8 Idaho National Laboratory, RELAP5-3D© Code Manual Volume I: Code Structure, System Models, And Solution Methods, INEEL-EXT-98-00834 Revision 2.4, June 2005.

15.0-9 K.F. Eckerman et al., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factor for Inhalation, Submersion, and Ingestion, Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

15.0-10 K.F. Eckerman and J.C. Ryman, External Exposure to Radionuclide in Air, Water and Soil, Federal Guidance Report 12, EPA-402-R-93-801, Environmental Protection Agency, 1988.

15.0-11 Single Failure Criterion. SECY-77-439, August 1977.

15.0-12 Electric Power Systems, 10CFR Part 50, Appendix A, General Design Criterion 17, “.

15.0-13 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.2 (Proprietary) and MUAP-07008-NP Rev.2 (Non-Proprietary), ~~May 2007~~July 2010.

15.0-14 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), October 2010~~July 2007~~.

15.0-15 ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, Approved August 29, 1979.

15.0-16 S.L. Humphreys et al., RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose estimation, NUREG/CR-6604, U.S. Nuclear Regulatory Commission, April 1998.

15.0-17 International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, 1979.

15.0-18 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P (Proprietary) and MUAP-07011-NP (Non- Proprietary), July 2007.

15.1.6 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.1.7 References

15.1-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.1-2 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), ~~July~~ October ~~2007~~2010.

15.1-3 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.1-4 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206, June 2007.

15.1-5 Moody, F. J., Maximum Flow Rate of Single Component, Two-phase Mixture, Journal of Heat Transfer, Trans. of the ASME, No.1, Feb., 1965.

15.1-6 Thermal Design Methodology, MUAP-07009-P (Proprietary) and, MUAP-07009-NP (Non-Proprietary), May 2007.

15.2.9 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.2.10 References

15.2-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.2-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.2-3 American Nuclear Society (ANS) 5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, Approved August 29, 1979.

15.2-4 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), ~~July~~ October ~~2007~~2010.

15.3.5 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.3.6 References

15.3-1 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), ~~July~~ October ~~2007~~2010.

15.3-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.3-3 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.3-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

- The primary leakage to containment which then leaks via the assumed design basis containment leakage.
- Engineered safety features (ESF) system leakage from manual actuation.

Figure 15A-5 depicts the leakage sources to the environment modeled in the dose computation.

Additionally, radionuclide decay of the nuclides is credited prior to release to the environment. After release, no decay is credited.

15.4.8.5.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are itemized in Table 15.4.8-3, Tables 15.0-8 through 15.0-10 and Tables 15.0-12 through 15.0-14.

The rod ejection accident releases iodines, alkali metals, and noble gases. In the reactor coolant, equilibrium iodine levels are assumed to exist along with the noble gas and alkali metal concentrations from the allowable design fuel defect. When compared to the releases from the fuel after the ejection event, the pre-existing activities are minor.

The total fission-product gap fraction available for release following any reactivity-initiated accident includes the steady-state gap inventory (present prior to the event) plus any fission gas released during the event.

The steady-state releases from the gap between the cladding and fuel is based on RG 1.183 (Ref. 15.4-4). The fission product gap inventory is increased to 10% of the iodines and noble gases, and 12% of the alkali metals. It is conservatively assumed that the failed fuel rods are operating at levels above the core average, and the releases are increased by the appropriate radial peaking factor.

In addition, transient fission gas release from the fuel rod is considered to be based on SRP 4.2, Appendix B (Ref. 15.4-7). The transient release is assumed to be 11% of the iodines and noble gases.

For this analysis, it is conservatively assumed that the fraction of melted fuel is 0.25%. For the melted fuel, 100% of the noble gases and 50% of the iodines and alkali metals are assumed to be released to the reactor coolant.

It is further assumed that the secondary concentration due to pre-existing primary-to-secondary leakage is 10% of the maximum primary equilibrium concentration for the iodines and alkali metals. The primary-to-secondary leakage of 600 gpd is high compared to the actual leakage rate.

It is assumed that leakage from ESF system leakage occurs when operator starts manual actuation of containment spray systems. With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to

Table 15.4.8-3
Parameters Used in Evaluating the Radiological Consequences of the Rod Ejection Accident (Sheet 1 of 3)

Parameter	Value
Core thermal power level ~~(%)~~(MWt)	4540 (2% above the design core thermal power)
Initial reactor (primary) coolant activity (from rods leaking prior to transient)	
Iodine	Initial concentration equal to the 1.0 μCi/g DE I-131 in the reactor coolant. (See Table 15.0-10.)
Alkali metals	Based on 1% fuel defect (See Table 11.1-2.)
Noble gases	300 μCi/g DE Xe-133 (See Table 15.0-12.)
Initial secondary coolant activity (from rods leaking prior to transient)	
Secondary system initial iodine and alkali metal concentration	Based on 10% of reactor coolant concentration
Source Term	
Core Activity	See Table 15.0-14.
Fraction of core inventory released from failed rods	
Fraction of fuel rods assumed to fail during transient (%)	10
Radial power peaking factor (to calculate fraction of total inventory in failed rods)	1.78
Iodine fission product gap fraction Alkali metal fission product gap fraction Noble gas fission product gap fraction	See Table 15.0-8.
Transient release fraction from the fuel rods (%)	11
Fraction of melted fuel (%)	0.25
Fraction of activity released from melted fuel: Iodine and alkali metals (containment release case) (Secondary side release case) Noble gases	 0.25 0.5 1.0
Iodine chemical form released from the SGs Iodine chemical form released from the containment: Iodine chemical form released from the ESF:	elemental: 97%, organic: 3% elemental: 4.85%, organic: 0.15%, particulate: 95% elemental: 97%, organic: 3%

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to US-APWR.

15.4.10 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.4.11 References

15.4-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.4-2 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), ~~July~~ October ~~2007~~ 2010.

15.4-3 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.4-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.4-5 Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, NRC Regulatory Guide 1.77, May 1974.

15.4-6 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.2 (Proprietary) and, MUAP-07008-NP Rev.2 (Non-Proprietary), ~~May 2007~~ July 2010.

15.4-7 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 4.2 Revision 3, Appendix B, March 2007.

15.5.3 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.5.4 References

15.5-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.5-2 Non-LOCA Methodology, MUAP-07010-P Rev.1 (Proprietary) and MUAP-07010-NP Rev.1 (Non-Proprietary), ~~July~~ October ~~2007~~2010.

For the CVCS letdown break outside the containment, the flow leaving the containment is at a low temperature, because the flow stream passes through the CVCS heat exchangers. This event is not analyzed, because the postulated sample line break is more limiting.

15.6.2.3 Core and System Performance

The size of a sample line is smaller than the break size corresponding to the makeup flow rate. Thus, the pressurizer water level can be maintained for a break in that line. As the makeup water is sufficient to maintain pressurizer water level within its normal operational range, no fuel damage results from this transient. The transient is terminated when the operator isolates the break and performs an orderly shutdown.

15.6.2.4 Barrier Performance

The RCS pressure remains well below 110% of the design pressure. However, this event postulates that both the primary system and containment systems fail due to a failure of small lines carrying primary coolant outside containment.

15.6.2.5 Radiological Consequences

The radiological consequences evaluation for this event uses the alternative source term (AST) guidance documented in Reference 15.6-4.

The radiological consequences evaluation assumes the reactor has been operating ~~with~~ at the maximum allowable limit for reactor coolant concentration. The equilibrium concentrations assumed in the analysis are based on Technical Specification coolant concentration limits.

In addition, it is assumed that iodine spiking (concurrent iodine spikes) occurs at the time of the accident. The concurrent iodine spike case postulates the iodine release rate from the detective rods increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g dose equivalent (DE) I-131) specified in the Technical Specifications. This is consistent with the methodology given in Reference 15.6-4.

The activity released from the defect fuels is assumed to be released instantaneously and homogeneously through the reactor coolant. The reactor coolant which is spilled in the auxiliary building collects in the floor drain sumps before being pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant, which are the noble gases and iodines.

In the failure of the sample line, the reactor coolant that is spilled from the break is assumed to be at maximum normal RCS pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

- Upon indication of a sample line break, the operator takes action to isolate the break by closing the operable isolation valve for the damaged line (See Table 7.5-5). The operator is assumed to detect and isolate the break within 45 minutes.

- The only filtration system considered in the analysis which limits the consequences of the failure of small lines carrying primary coolant outside containment is the main control room (MCR) and the technical support center (TSC) heating, ventilation, and air conditioning (HVAC) system.

- The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from NRC Regulatory Guide 1.183 (Ref. 15.6-4).

15.6.2.5.3 Results

As shown in Table 15.6.2-2, the calculated TEDE doses are determined to be 1.5 rem at the EAB and 0.60 rem at the LPZ outer boundary.

These doses are less than 10% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34. The dose guideline is based on the acceptance criterion given in SRP 15.6.2.

The doses for the MCR and TSC for the failure of small lines carrying primary coolant outside containment are bounded by the MCR doses calculated for the loss-of-coolant accident (LOCA) event described in ~~subsection~~ Section 15.6.5.5.

15.6.2.6 Conclusions

A single charging pump provides enough makeup water to the bounding small line break size so that the pressurizer level can be maintained in its normal operational range throughout the transient. Therefore, no fuel damage results from this event.

The resultant doses are well within the guideline values of 10 CFR 50.34.

Table 15.6.2-1
Parameters Used in Evaluating the Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

Parameter	Value
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Reactor coolant iodine concentration	The reactor coolant iodine concentration is based on a concurrent iodine spike corresponding to 500 times the release rate of iodine at the equilibrium value (1 μCi/g DE I-131).
Reactor coolant noble gas concentration	300 μCi/g DE Xe-133
Break flow rate (gpm)	97 (at density of 62.4 lb/ft^3~~3~~)
Fraction of reactor coolant flashing	47% based on initial reactor coolant enthalpy at maximum normal RCS pressure and final reactor coolant enthalpy at atmospheric pressure
Duration of accident (min)	45
MCR and TSC Parameters	
Envelope volume	See Table 15.6.5-5.
Occupancy frequency	See Table 15.6.5-5.
Total unfiltered inleakage	See Table 15.6.5-5.
HVAC system	See Table 15.6.5-5.
χ/Q	See Table 15.0-13 and 15A-21.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.6.2-2
Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

Dose Location	TEDE Dose (rem)
EAB dose (0 to 2 hours)	1.5
LPZ boundary dose	0.60
MCR [*1]	0.21
TSC	Less than MCR LOCA dose

Note:
[*1]The direct radiation shine dose at the time of LOCA is added as a direct radiation shine dose.

initial conditions, power distributions, and global and local parameters for the DECLG break. Some bounding parameters are fixed and selected to obtain a conservative estimate of PCT. One such parameter is the containment pressure, which affects the PCT and contains an uncertainty, as described in Chapter 6, Section 6.2.

Applying the Wilks' equation (Ref. 15.6-26), it needs 59 ASTRUM runs to obtain the 95th percentile for one parameter (i.e. PCT) with 95% confidence. The number of runs (N) for three parameters (i.e., PCT, LMO and CWO) with 95th percentile and 95% confidence is 124, obtained using the following equation:

$$\beta \le 1-\sum_{k=0}^{2} {}_{N}C_{k}\,\alpha^{N-k}\left(1-\alpha\right)^{k}$$

where: α = 0.95 (95th percentile) β= 0.95 (95% confidence), k is the number of evaluation parameter, and N is the number of runs. The detail procedure to yield the 124 runs is described in the Topical Report (Ref.15.6-9) and Reference 15.6-15.

Applying ASTRUM to calculate the total uncertainty in the PCT and other parameters, all the uncertainty parameters are sampled simultaneously in random in the WCOBRA/TRAC runs. Local parameters are those that affect the local fuel response at the hot spot. The local uncertainty is incorporated in the HOTSPOT code (Ref.15.6-8) to evaluate the PCT.

15.6.5.3.1.2 Small Break LOCA Evaluation Model

The small break LOCA analysis is performed using the M-RELAP5 code (Ref. 15.6-14), a modified version of the RELAP5-3D, which has multi-dimensional thermal-hydraulics and kinetic modeling capability. One-dimensional modeling with M-RELAP5 is used for LOCAs with break sizes less than 1.0 ft^2.

The following modifications were made to the M-RELAP5 code to incorporate 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements that are also in accordance with the TMI Action Item II.K.3.30 and II.K.3.31.

- Addition of ANS-1971 x 1.2 fission product decay curve
- Addition of Baker-Just correlation (not steam-limited) for metal-water reaction rate calculations
- Addition of ZIRLO^{TM} burst model
- For choked-flow calculation, the Moody model (steam quality > 0.0~~1~~) and maximum of Henry-Fauske and Moody ~~the Henry-Fauske~~ models (steam quality < 0.0~~1~~) are incorporated to model the discharge
- Return to nucleate and transition boiling heat transfer modes are prevented for the initial blowdown phase

noding, time-step size and input sensitivity studies. The sensitivity analyses are performed by complying with the requirements set forth in 10 CFR Appendix K to Part 50 on ECCS Evaluation Models. The objective is performed to determine the effects of various modeling assumption on the calculated PCT, LMO and CWO. Three small break LOCA cases are reported in this section. They are as follows:

- 7.5-inch ~~upside~~ downside break, which is the limiting break for PCT during the loop-seal clearance phase.
- 1-ft^2 ~~upside~~ downside break, which is the limiting break for PCT during the boil-off phase.
- 3.4-inch break, which is a DVI line break, with only 1 train of SI system is assumed to operate.

The major plant parameters inputs used in the Appendix-K based small break LOCA analysis are listed in Table 15.6.5.2. The top-skew axial power shape is chosen because it provides the distribution of power versus core height that maximizes the PCT. Figure 15.6.5-13 shows the hot rod power shape used to conduct the small break LOCA analysis. The hot rod power shape considers the axial off-set limits of the core design, and is conservative compared to the limiting large-break LOCA power shape. The beginning of life (BOL) hot assembly burnup provides the maximum (conservative) initial stored energy in the fuel for the SBLOCA event. In addition, for the hot rod, an initial highest pellet temperature is also assumed for conservatism.

In addition to the conditions in Table 15.6.5-2, the following conditions are also applicable to the SBLOCA.

- The limiting single failure in the small break LOCA analysis is assumed, which is the loss of one ECCS train, with one additional train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- LOOP is assumed to occur simultaneously with the reactor trip, resulting in the delay of SI pumps and EFWS operations. RCP trip is assumed to occur 3 seconds after the reactor trip, as described in Section 15.0.0.7.
- Shutdown reactivities resulting from fuel temperature and void are given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors as shown in Table 15.6.5-2. Control rod insertion is considered to occur and assumed in the analysis.

15.6.5.3.2.3 Post-LOCA Long Term Cooling

The major input parameters used in the long term cooling evaluation are listed in Table 15.6.5-3. In this evaluation, atmospheric pressure is assumed as the lowest

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses are carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling is achieved.

Based on the analysis, the application of ASTRUM for the best-estimate analysis of the large break LOCA shows that the acceptance criteria of 10 CFR 50.46 are satisfied for the US-APWR. In addition, it is confirmed that 2 (two) safety injection trains are capable of satisfying the design cooling function for any large break LOCA, assuming a single failure of one train, and another train out of service for maintenance.

15.6.5.3.3.2 Small Break LOCA Analysis Results

Details for the limiting small break LOCA are presented in this section. The results for other cases are documented in detailed in Technical Report (Ref. 15.6-16).

Results of 7.5-inch Small Break LOCA Analysis

The sequence of events for the 7.5-inch small break LOCA is presented in Table 15.6.5-9. Depressurization of the RCS (Figure 15.6.5-14) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint of 1860 psia is reached. The reactor trips at 9.3 seconds, then the power decreases (Figure 15.6.5-15). Control rod insertion starts at 11 seconds, which is concurrent with the turbine trip and main steam isolation. Voiding in the core also causes the reactor power to decrease.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-16. During the earlier part of the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. The ECCS actuation signal occurs at 12 seconds when the low pressurizer pressure setpoint is reached. This is immediately followed by the RCPs trip just before 13 seconds. The main feedwater flow is isolated at 17 seconds. To limit the pressure build up in the secondary system, the main steam safety valves open at ~~81~~ 78 seconds. The upper region of the core begins to uncover at ~~122~~ 124 seconds. Figure 15.6.5-17 shows the accumulator and safety injection mass flowrates transient. The HHIS begins to inject borated water to the reactor core at 130 seconds. The accumulators begin injecting borated water into the cold-leg at about 300 seconds.

As a result of the loop-seal clearance, the core is recovered at ~~142~~ 141 seconds. Figure 15.6.5-18 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level are shown in Figures 15.6.5-19 and 15.6.5-20, respectively.

Figure 15.6.5-21 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. The PCT of ~~773~~761°F occurs at ~~136~~ 137 seconds. This figure demonstrates that the PCT is substantially lower than 2200°F.

Figure 15.6.5-22 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-10 presents the calculated PCT, LMO, and CWO results for the limiting 7.5-inch small break LOCA. This case is the limiting break for PCT during the loop-seal clearance phase.

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F. The PCT of ~~773~~761°F presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. The maximum core wide cladding oxidation is lower than 0.2 % as presented in Table 15.6.5-10 in compliance with regulatory limit.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met since the PCT does not exceed 2200°F. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of 1-ft^2 Small Break LOCA Analysis

The sequence of events for the 1-ft^2 break, which is a 13.5-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This is the limiting break for PCT during the boil-off phase.

Figure 15.6.5-23 depicts the pressure transient in the pressurizer. Depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated at 6.9 seconds when the low pressurizer pressure setpoint is reached. LOOP is assumed at the same time with the reactor trip. The reactor power then decreases (Figure 15.6.5-24) following the reactor trip. Control rod insertion and main steam flow isolation occur at 8.7 seconds. The RCPs trip at 9.9 seconds, indicating 3 seconds delay from the reactor trip. Main feedwater flow is isolated at 15 seconds. Because secondary system pressure build up does not occur, the main steam safety valves remain closed.

The liquid and vapor discharges from the break are shown in Figure 15.6.5-25. Early in the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. Upward flow through the core is maintained. However, the flow rate is not sufficient to prevent partial uncovery in the core.

The ECCS actuation signal is generated when the low pressurizer pressure setpoint is reached at 8 seconds.

Figure 15.6.5-26 shows the accumulator and safety injection mass flow rates. The accumulators begin injecting borated water into the cold-leg at ~~90~~ 89 seconds. The HHIS begins to inject borated water to the reactor core at 126 seconds. As a result of ECCS injection, the mass inventory is recovered. Figure 15.6.5-27 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level transients are shown in Figures 15.6.5-28 and 15.6.5-29, respectively. Figure 15.6.5-30 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and for the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT of ~~1323~~1302°F occurs at ~~169~~ 161 seconds. The PCT is significantly lower than 2200°F.

Figure 15.6.5-31 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-12 presents the 1-ft^2 ~~upside~~ downside break, which is a 13.5-inch equivalent diameter small break LOCA.

1. The PCT of ~~1323~~1302°F presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

2. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

3. The maximum core wide cladding oxidation is lower than 0.2% as presented in Table 15.6.5-12, in compliance with regulatory limit.

4. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of the DVI-Line Small Break LOCA Analysis

The sequence of events for the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This case assumes the injection of only one SI pump.

Depressurization of the RCS (Figure 15.6.5-32) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint is reached at 26 seconds. The reactor power then decreases (Figure 15.6.5-33) following the reactor trip. Control rod insertion starts at 28 seconds, simultaneous with the turbine trip and main steam isolation. The RCP trips at 29 seconds, which is 3 seconds after the reactor trip.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-34. ~~Downward flow does not occur in this particular case. Upward flow through the core is maintained. The core flow is sufficient to prevent any uncovery of the core.~~

The ECCS actuation signal is initiated when the low pressurizer pressure setpoint of 1760 psia is attained at 35 seconds. In this case, the HHIS alone provides the core cooling function. Figure 15.6.5-35 shows the accumulator and safety injection mass flow rates. Figure 15.6.5-36 shows that the RCS inventory increases. The downcomer liquid collapsed level transient and core/upper plenum liquid collapsed level transient are shown in Figures 15.6.5-37 and 15.6.5-38, respectively.

Figure 15.6.5-39 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT ~~does not~~of 789°F occurs at 1505 seconds.~~in the DVI-line break, indicating that the core keeps covered throughout the transient.~~ The PCT is significantly lower than 2200°F.

Figure 15.6.5-40 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-14 presents the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA.

1. The PCT of 789°F presented in Table 15.6.5-14 indicates that this regulatory limit has been met.~~For the DVI-line break, no heatup occurs. This obviously demonstrates that the regulatory limit has been met.~~

Table 15.6.5-9

Sequence of Events for 7.5-inch Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	9.3
Control rod insertion starts	11.1
Main steam isolation	11.1
ECCS actuation signal	11.~~9~~8
RCP trip	12.3
Main feedwater isolation	17.3
Main steam safety valve open	~~81~~78
Emergency Power Source initiates	115
Core upper region uncovery	~~122~~124
High Head Injection System begins	130
Peak Cladding Temperature occurs	~~136~~137
Core upper region recovery	~~142~~141
Emergency feedwater flow begins	145
Accumulator injection begins	~~299~~317

Table 15.6.5-10

Core Performance Results for 7.5-inch Small Break LOCA

	Values
Peak Cladding Temperature (°F)	~~773~~761
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-11

Sequence of Events for 1-ft^2 Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	6.9
ECCS actuation signal	8.3
Control rod insertion starts	8.7
Main steam isolation	8.7
RCP trip	9.9
Main feedwater isolation	14.9
Main steam safety valve open	not actuated
Accumulator injection begins	~~90~~89
Core upper region uncovery	96
Emergency Power Source initiates	111
High Head Injection System begins	126
Emergency feedwater flow begins	141
Peak Cladding Temperature occurs	~~169~~161
Core upper region recovery	~~339~~356

Table 15.6.5-12

Core Performance Results for 1-ft^2 Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	~~1323~~1302
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-13

Sequence of Events for DVI-line Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip, (LOOP is assumed)	~~25.9~~25.8
Control rod insertion starts	~~27.7~~27.6
Main steam isolation	~~27.7~~27.6
RCP trip	~~28.9~~28.8
Main feedwater isolation	~~33.9~~33.8
ECCS actuation signal	35.4
Main steam safety valve open	57
Emergency Power Source initiates	138
High Head Injection System begins	153
Emergency feedwater flow begins	168
Core upper region uncovery	~~not occur~~1256
Peak Cladding Temperature	~~lower than the initial value~~1505
Core upper region recovery	~~N/A~~1856

Table 15.6.5-14

Core Performance Results for DVI-line Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	~~lower than the initial value~~789
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	N/A

Table 15.6.5-15

Spectrum of Peak Cladding Temperatures for Small Break LOCA

Break size and orientation	PCT
1-ft^2 at cold leg (bottom)	~~1174~~1302°F
13-inch at cold leg (bottom)	~~1154~~1250°F
12-inch at cold leg (bottom)	~~938~~1220°F
11-inch at cold leg (bottom)	lower than the initial temperature
10-inch at cold leg (bottom)	lower than the initial temperature
9-inch at cold leg (bottom)	lower than the initial temperature
8-inch at cold leg (bottom)	~~lower than the initial temperature~~701°F
7.5-inch at cold leg (bottom)	761°F
7-inch at cold leg (bottom)	~~756~~715°F
6.5-inch at cold leg (bottom)	~~lower than the initial temperature~~691°F
6-inch at cold leg (bottom)	~~lower than the initial temperature~~708°F
5-inch at cold leg (bottom)	lower than the initial temperature
4-inch at cold leg (bottom)	lower than the initial temperature
3-inch at cold leg (bottom)	lower than the initial temperature
2-inch at cold leg (bottom)	lower than the initial temperature
1-inch at cold leg (bottom)	lower than the initial temperature

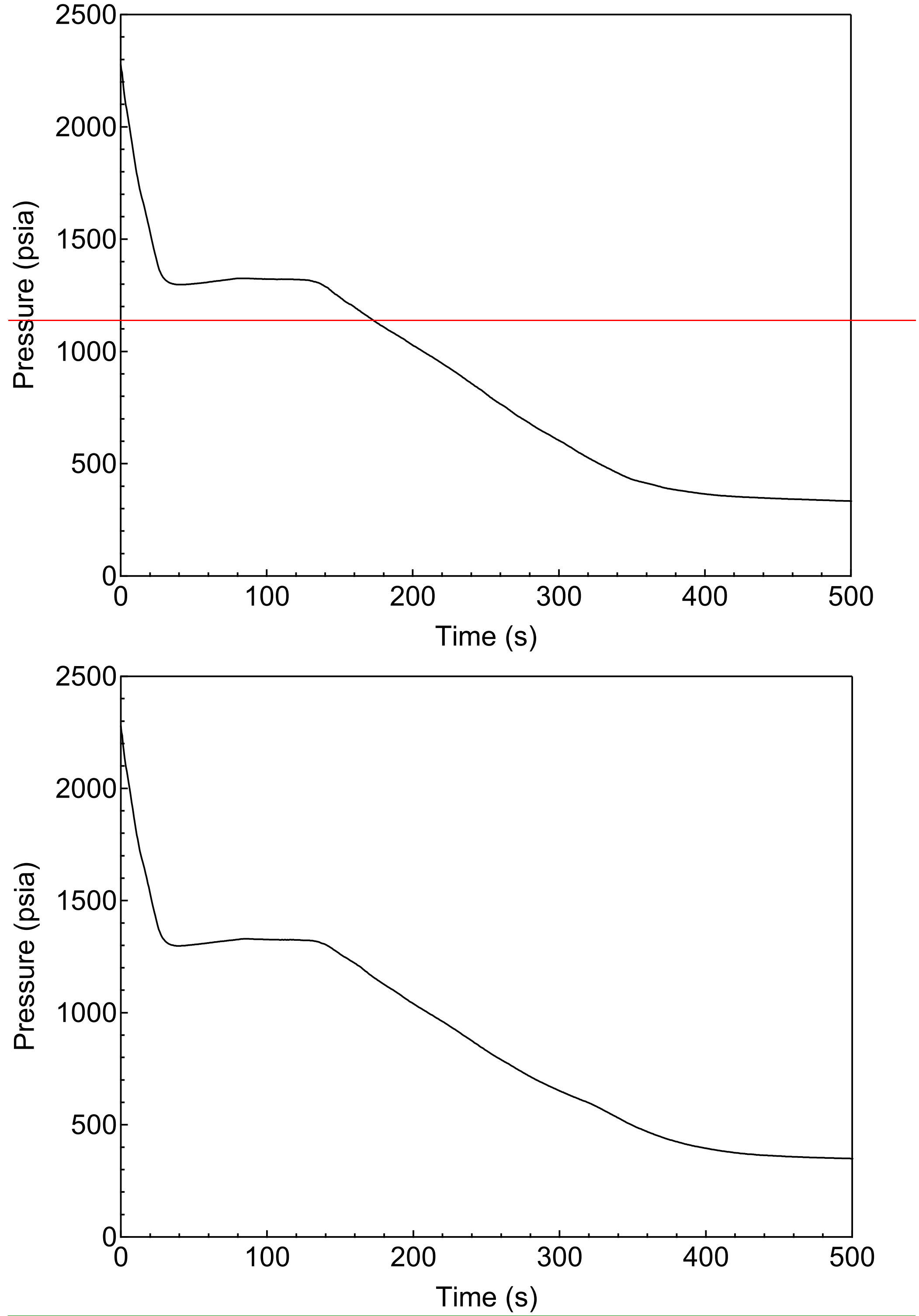
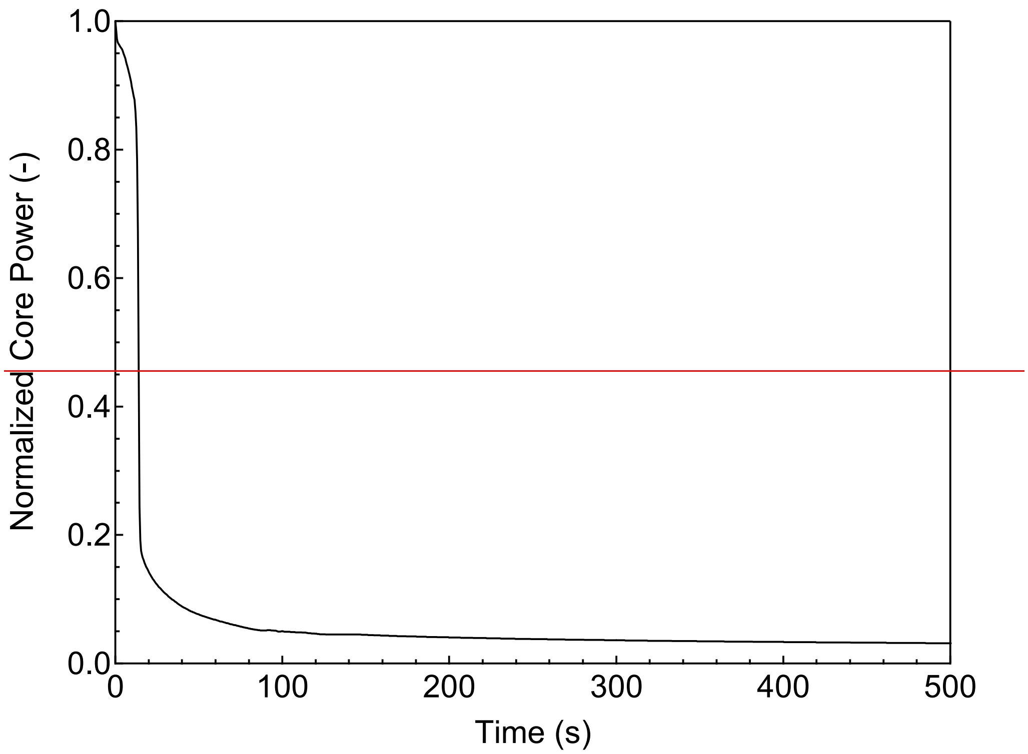


Figure 15.6.5-14 RCS (Pressurizer) Pressure Transient for 7.5-inch Small Break LOCA



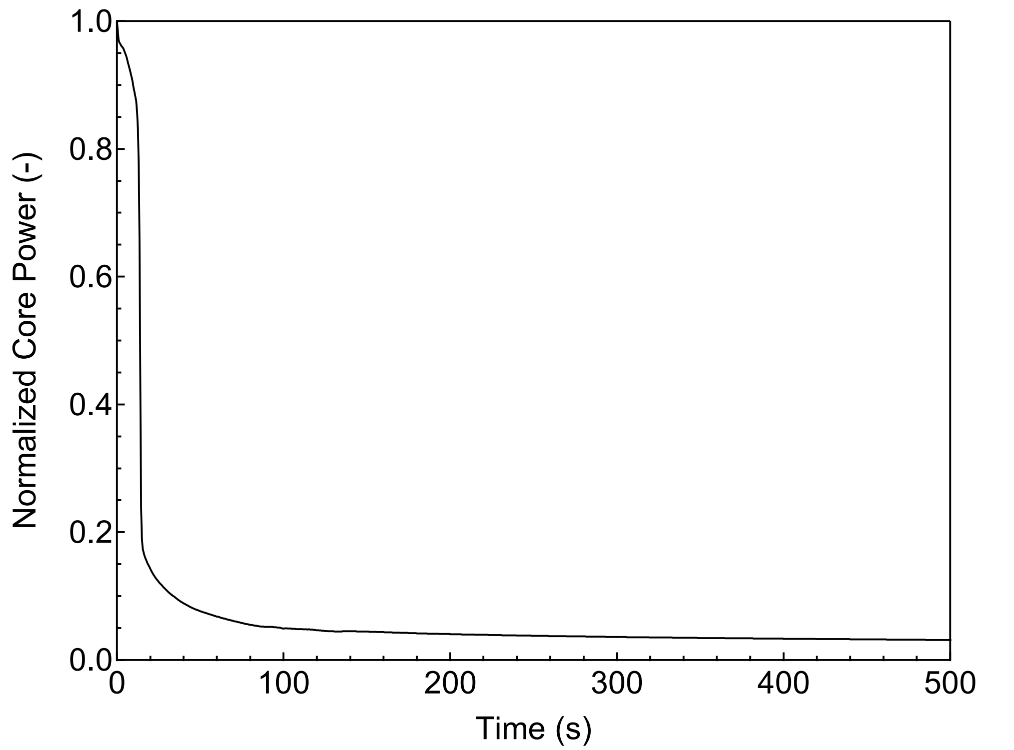


Figure 15.6.5-15 Normalized Core Power for 7.5-inch Small Break LOCA

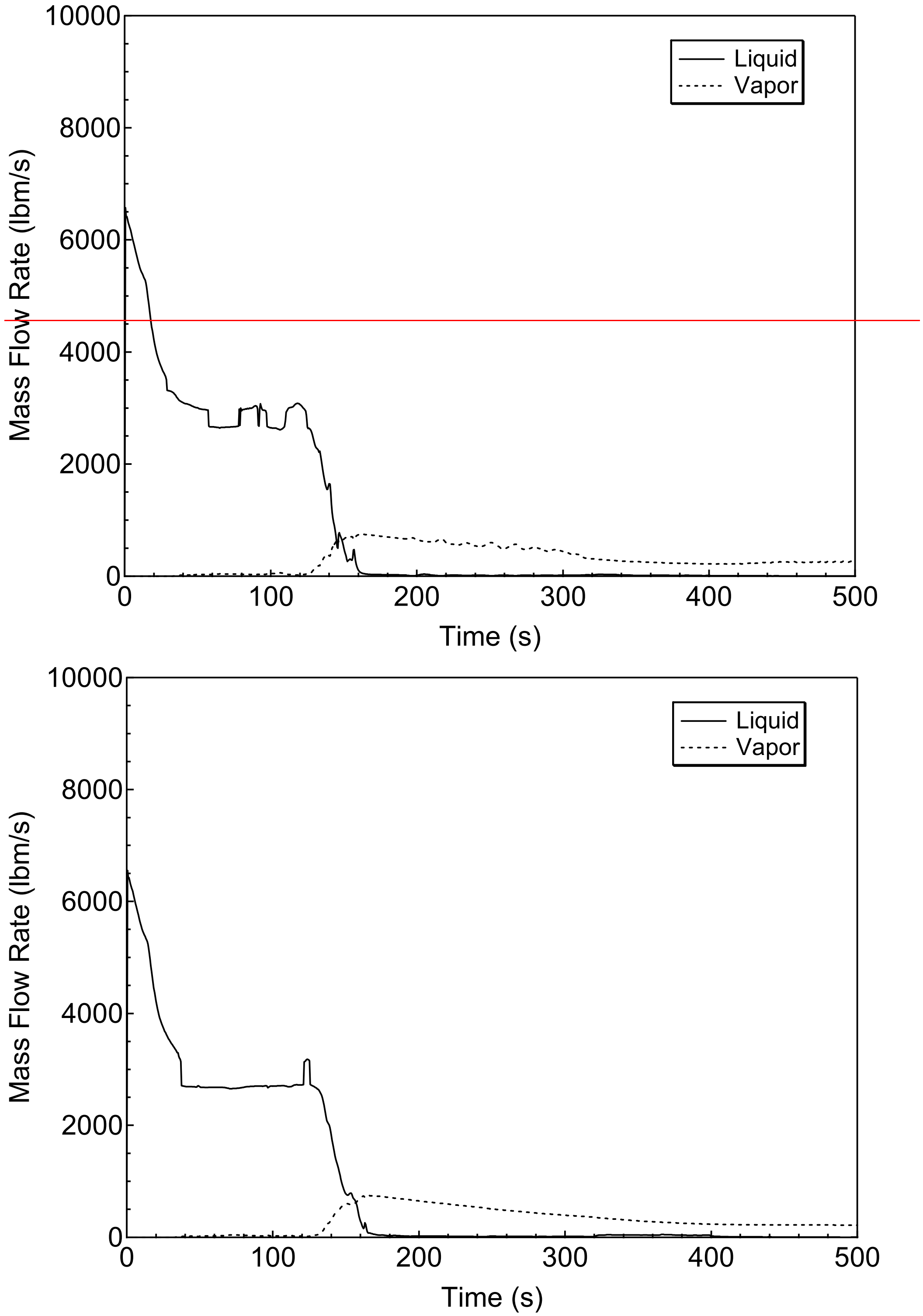


Figure 15.6.5-16 Liquid and Vapor Discharges through the Break for 7.5-inch Small Break LOCA

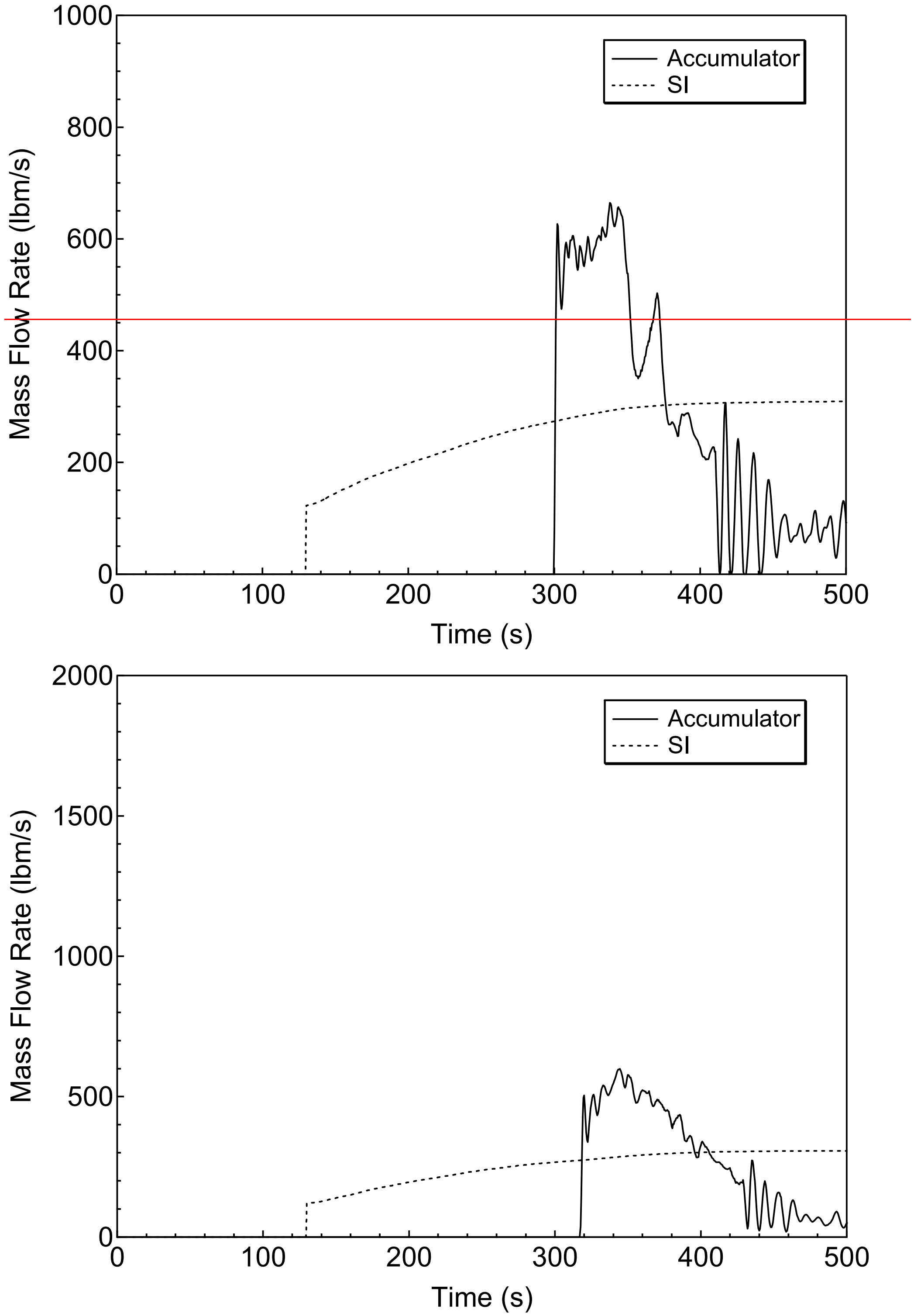


Figure 15.6.5-17 Accumulator and Safety Injection Mass Flowrates for 7.5-inch Small Break LOCA

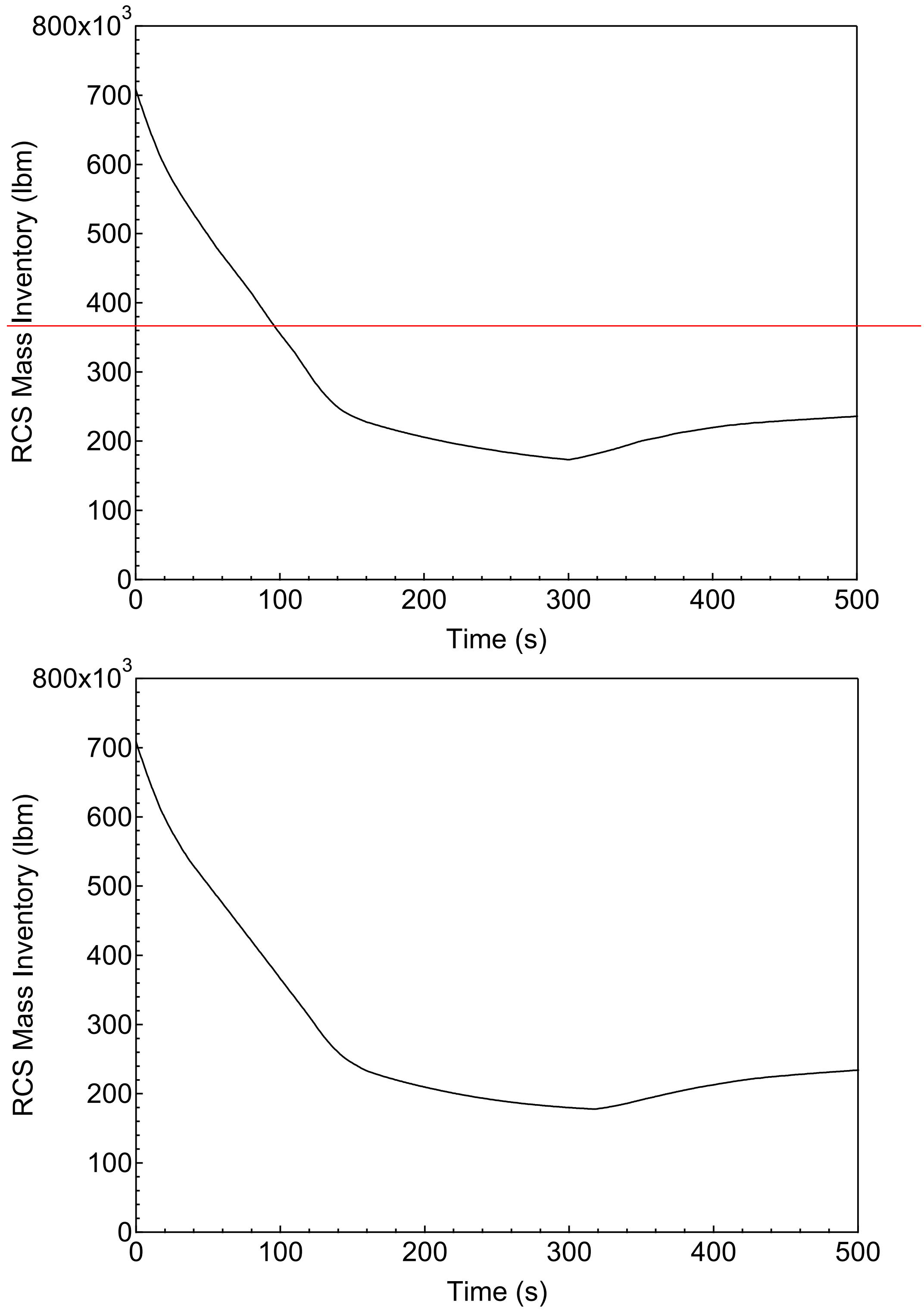
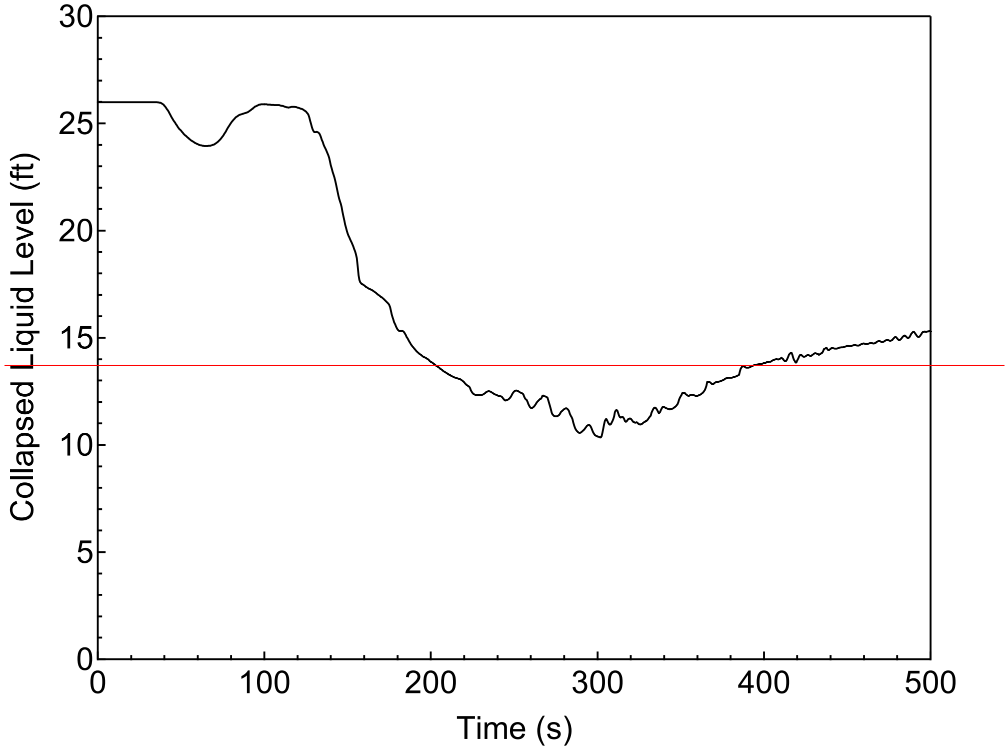


Figure 15.6.5-18 RCS Mass Inventory for 7.5-inch Small Break LOCA



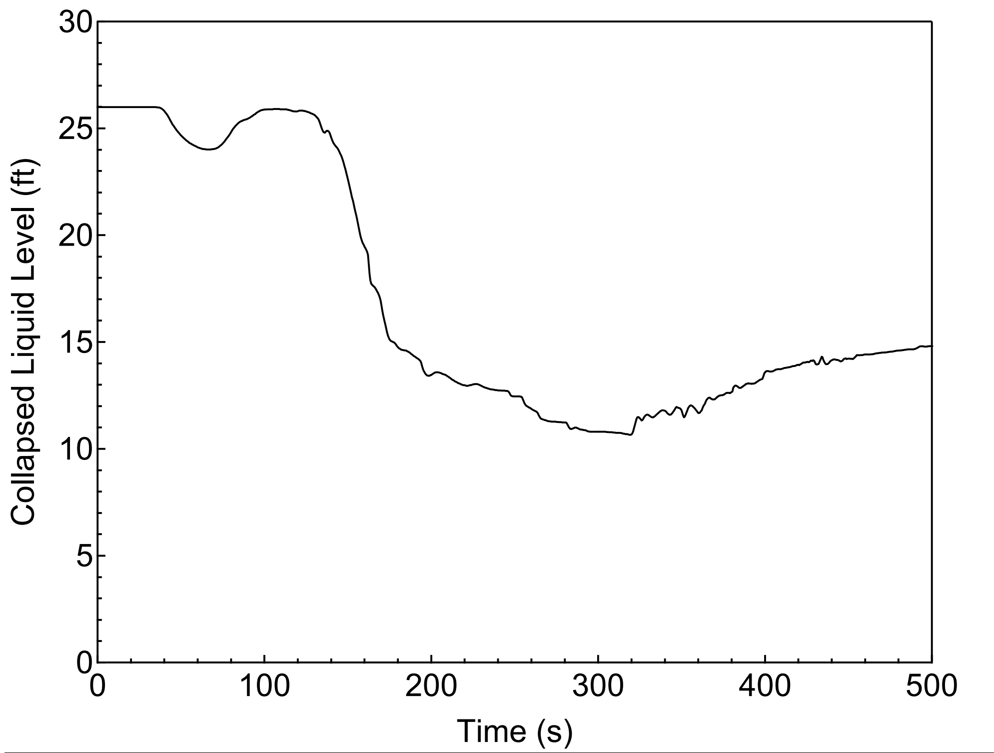
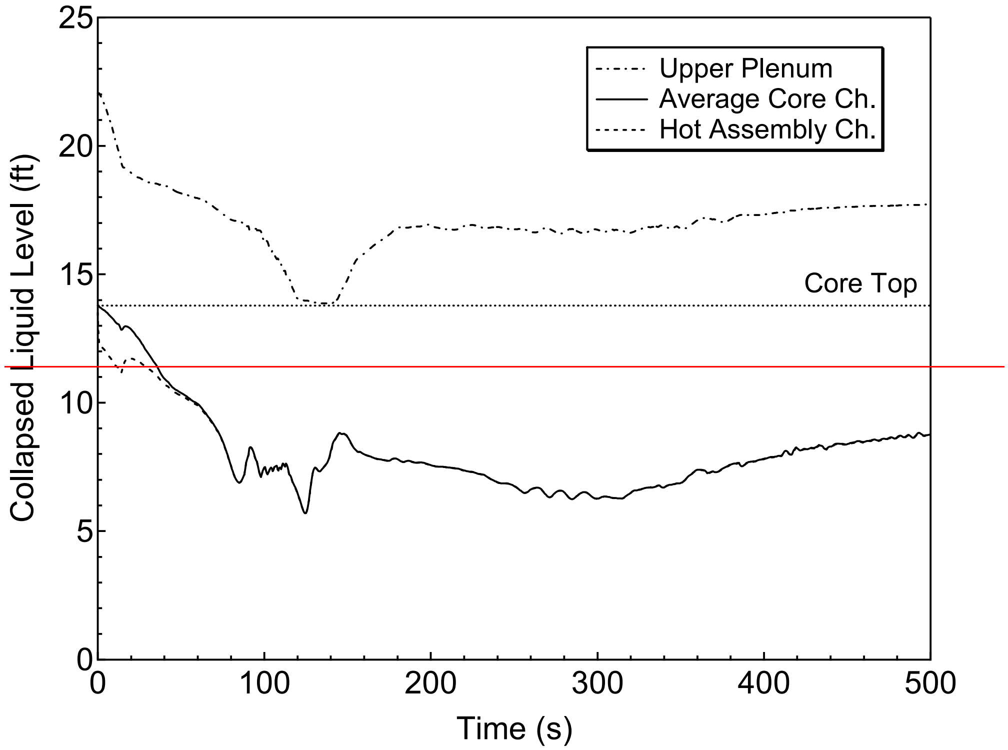


Figure 15.6.5-19 Downcomer Collapsed Level for 7.5-inch Small Break LOCA



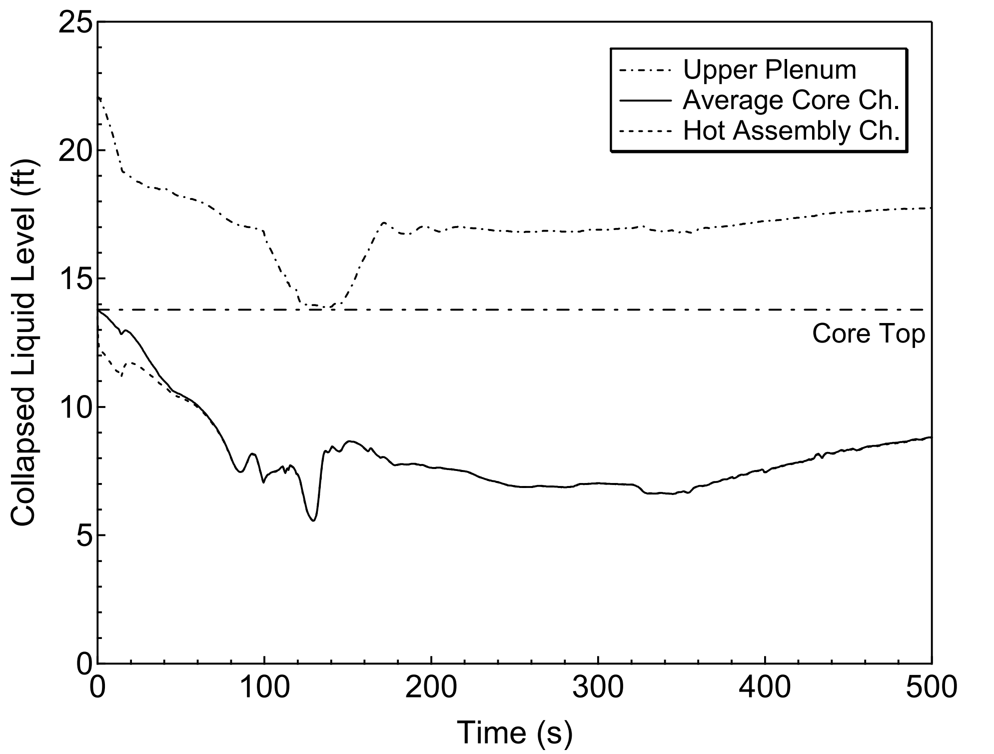


Figure 15.6.5-20 Core/Upper Plenum Collapsed Level for 7.5-inch Small Break LOCA

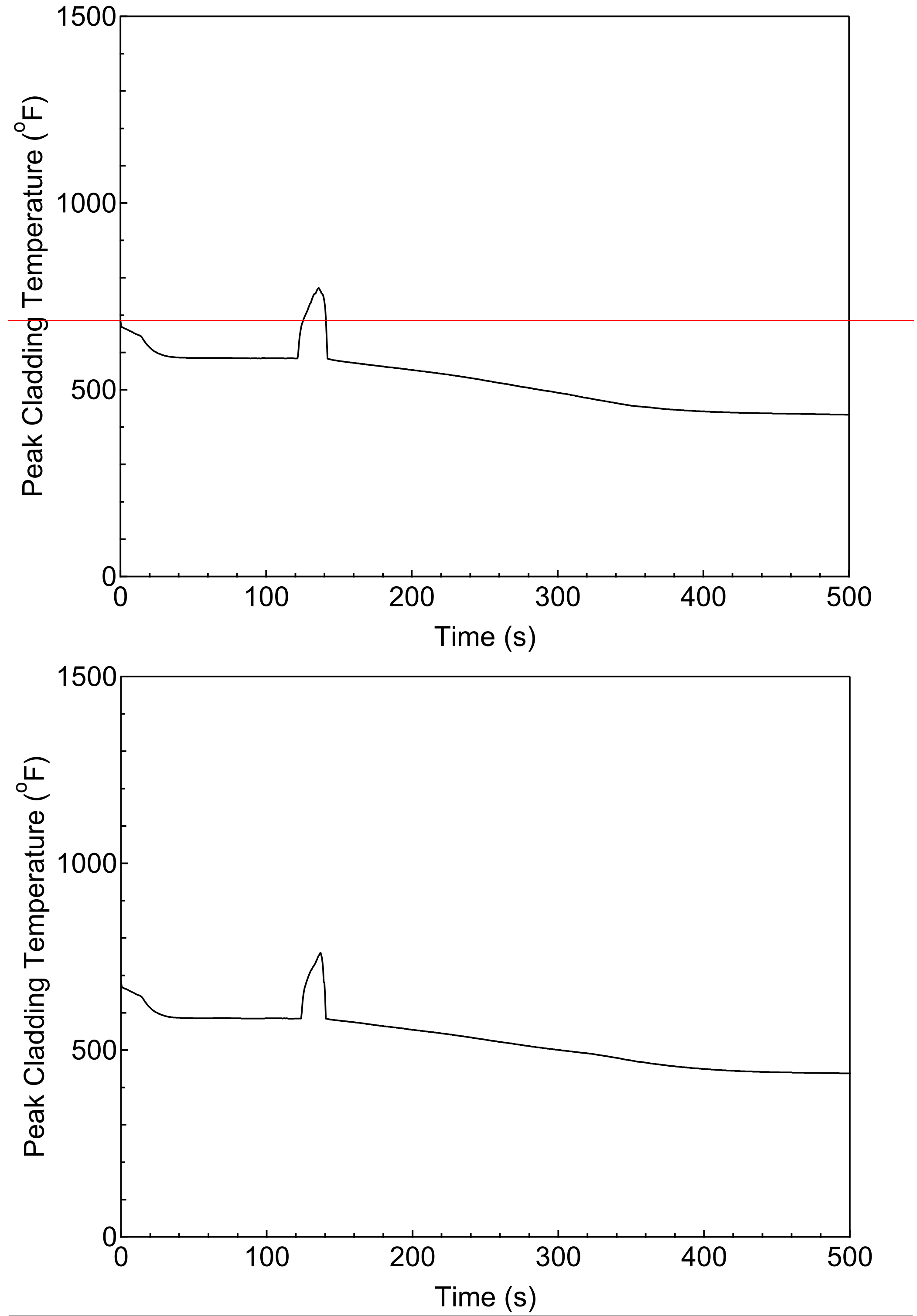


Figure 15.6.5-21 PCT at All Elevations for Hot Rod in Hot Assembly for 7.5-inch Small Break LOCA

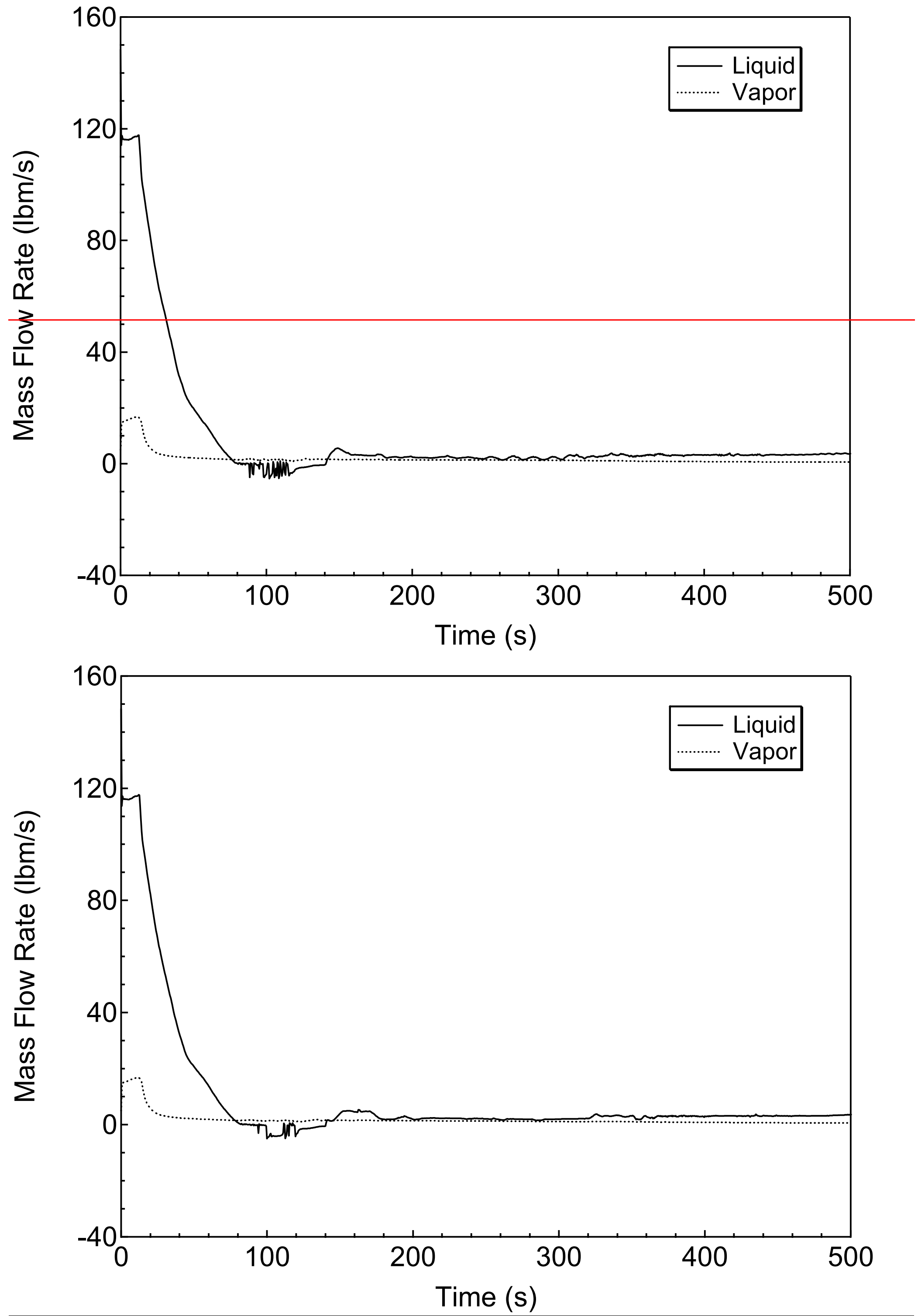


Figure 15.6.5-22 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 7.5-inch Small Break LOCA

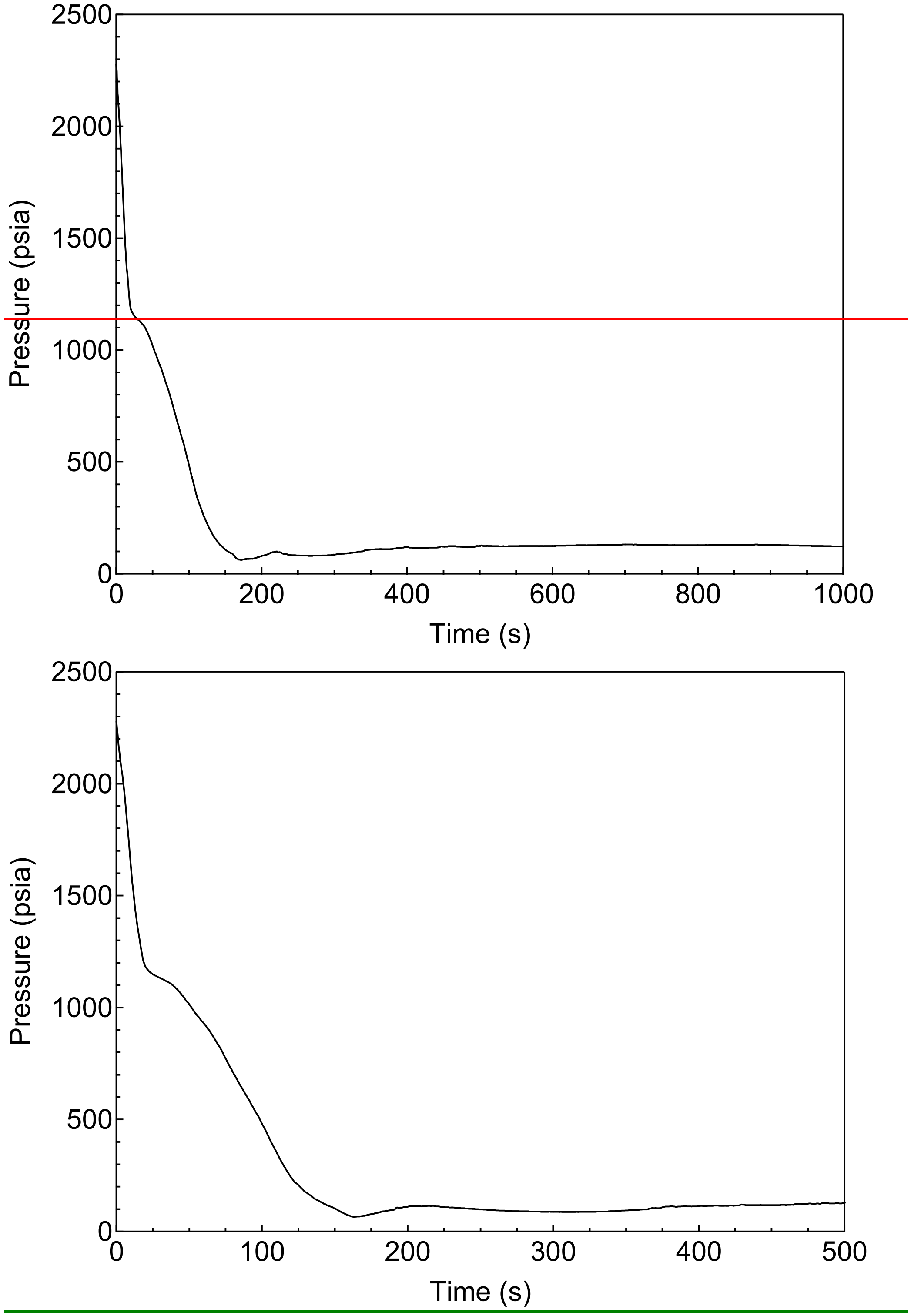
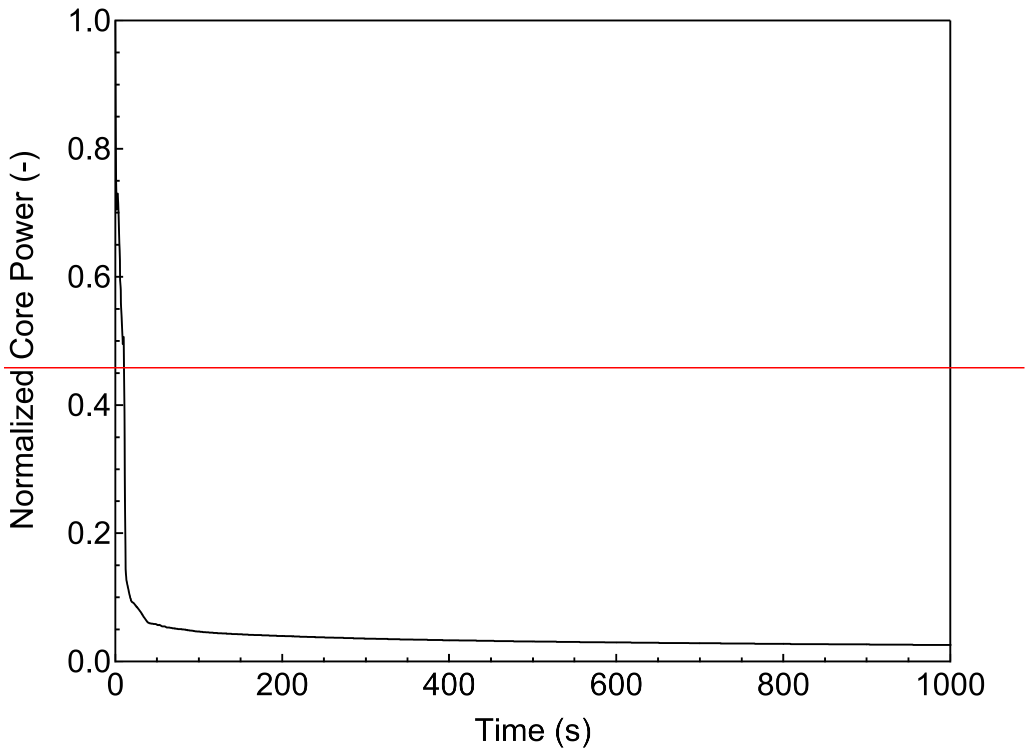


Figure 15.6.5-23 RCS (Pressurizer) Pressure Transient for 1-ft^2 Small Break LOCA



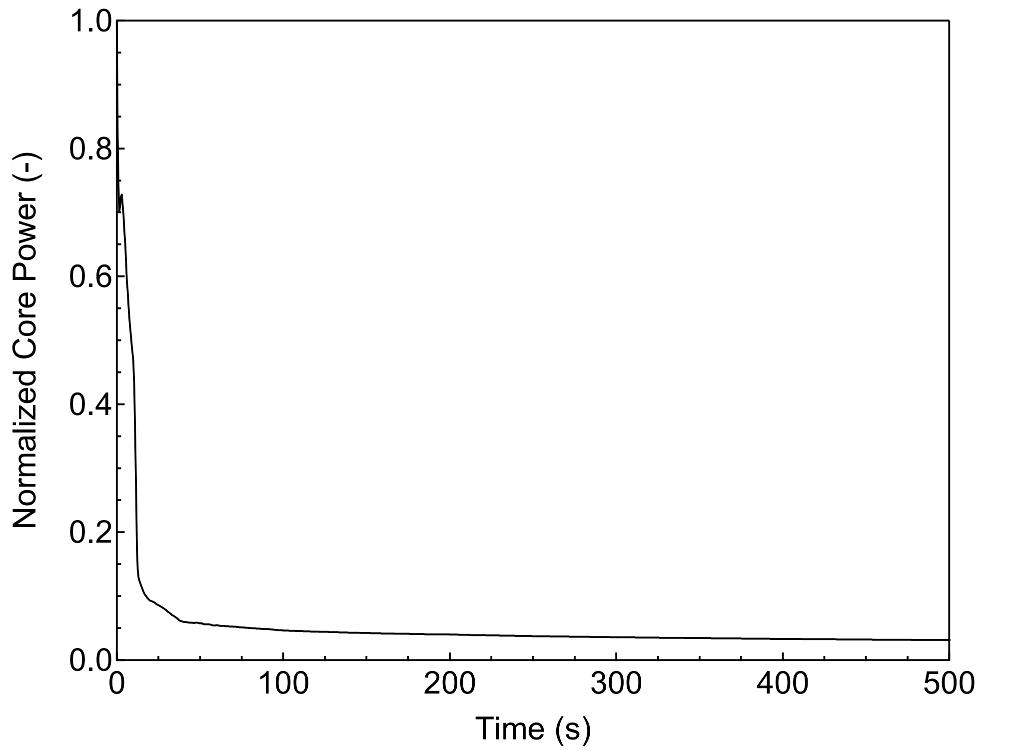


Figure 15.6.5-24 Normalized Core Power for 1-ft^2 Small Break LOCA

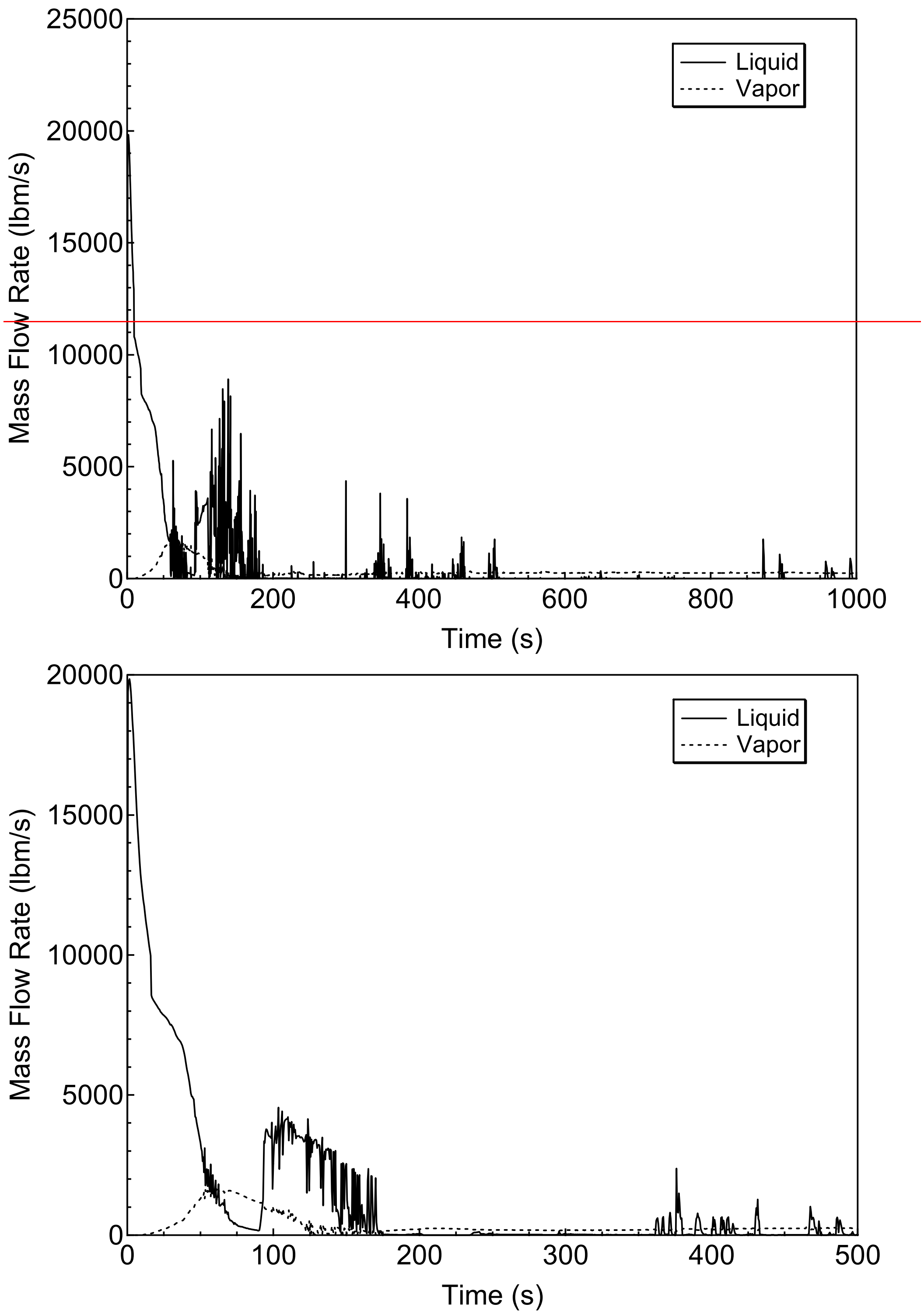


Figure 15.6.5-25 Liquid and Vapor Discharges through the Break for 1-ft^2 Small Break LOCA

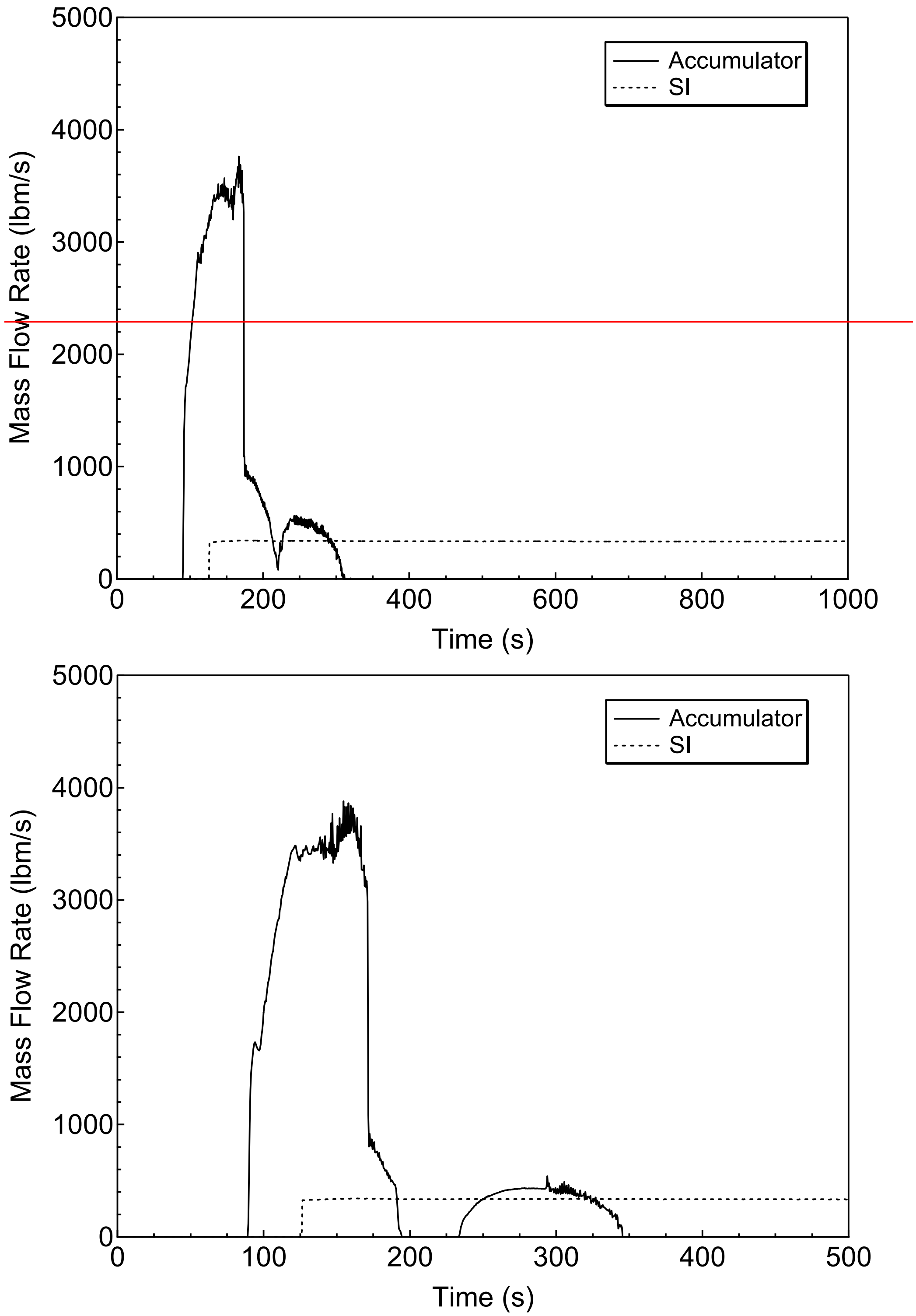


Figure 15.6.5-26 Accumulator and Safety Injection Mass Flowrates for 1-ft^2 Small Break LOCA

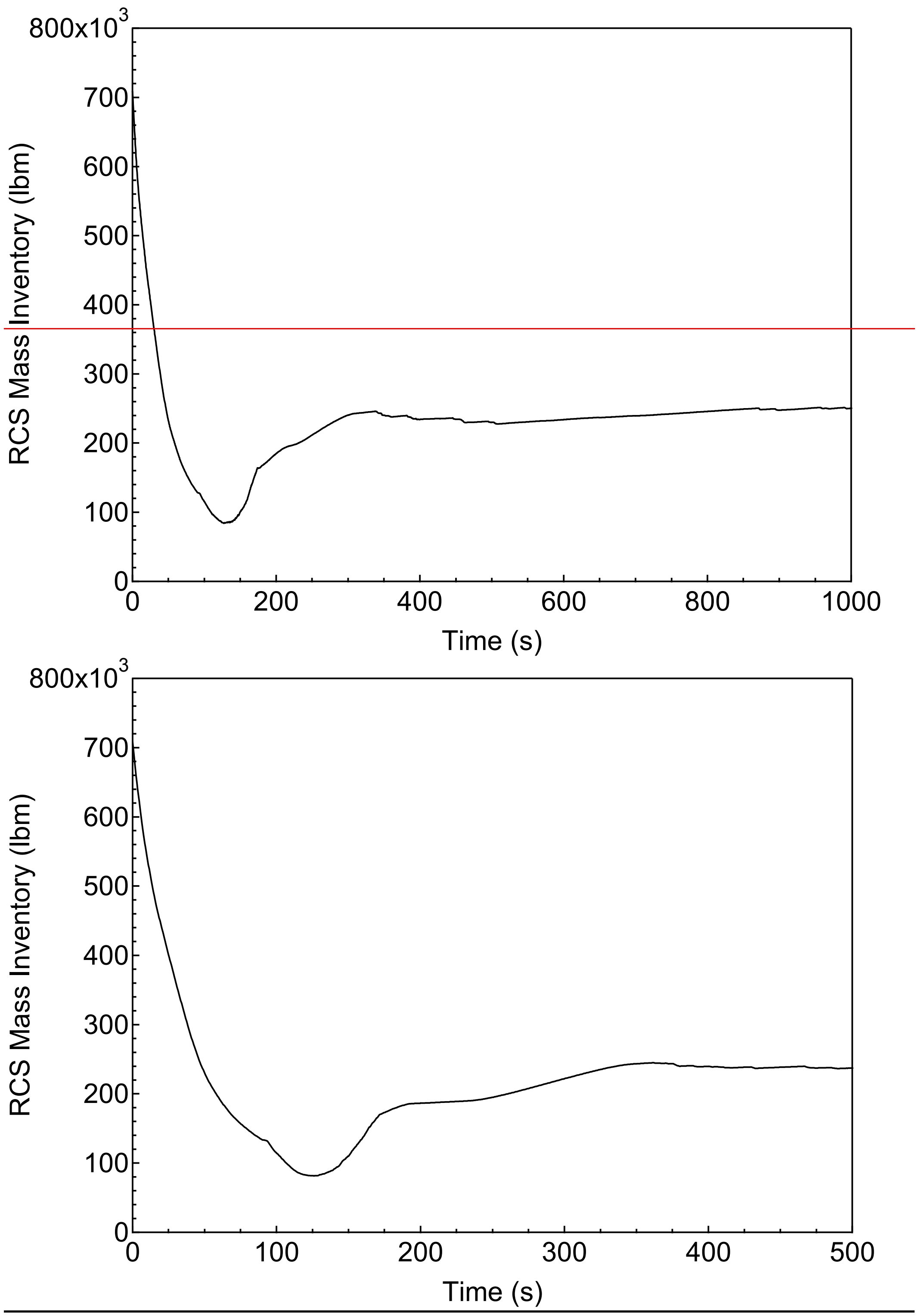


Figure 15.6.5-27 RCS Mass Inventory for 1-ft^2 Small Break LOCA

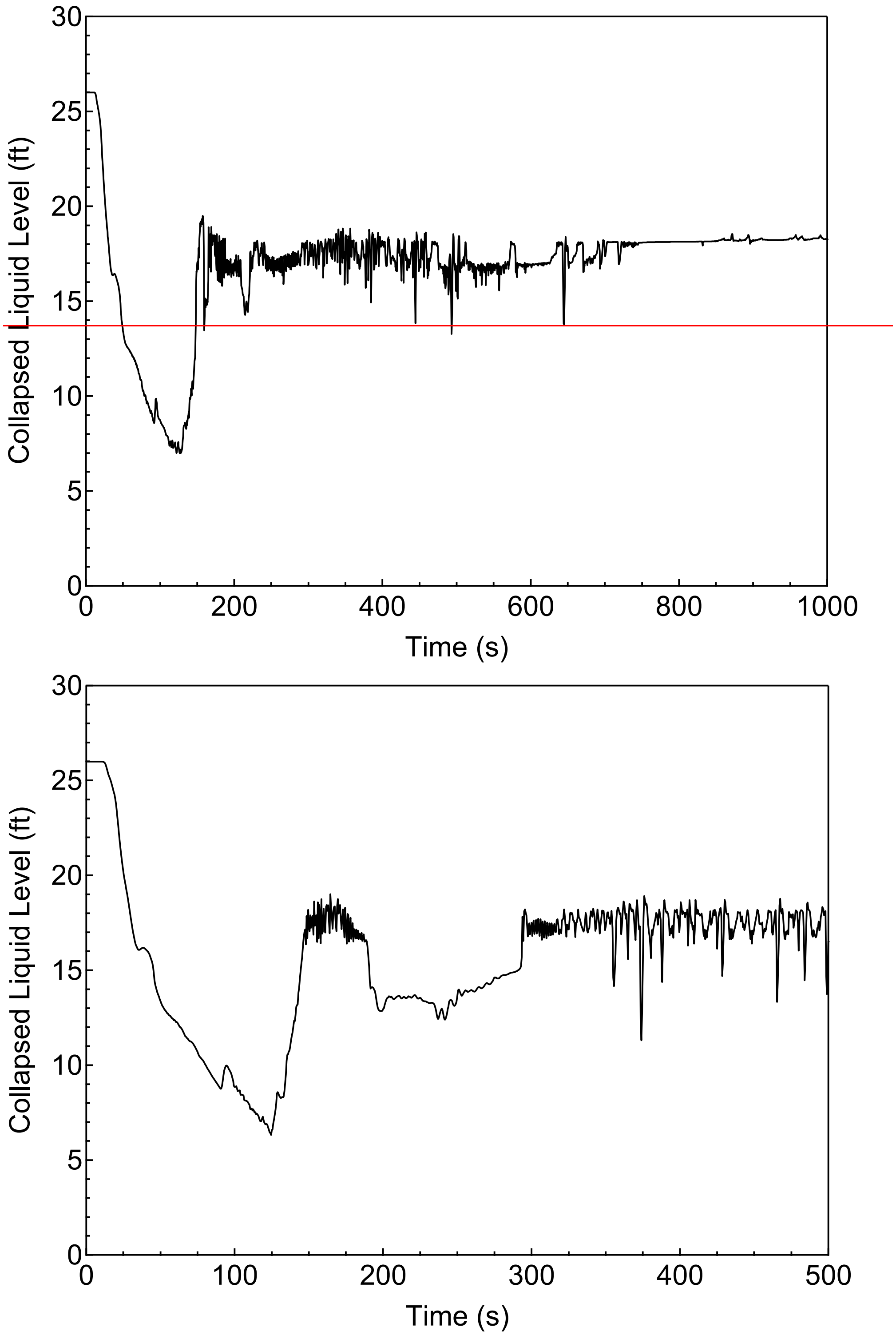
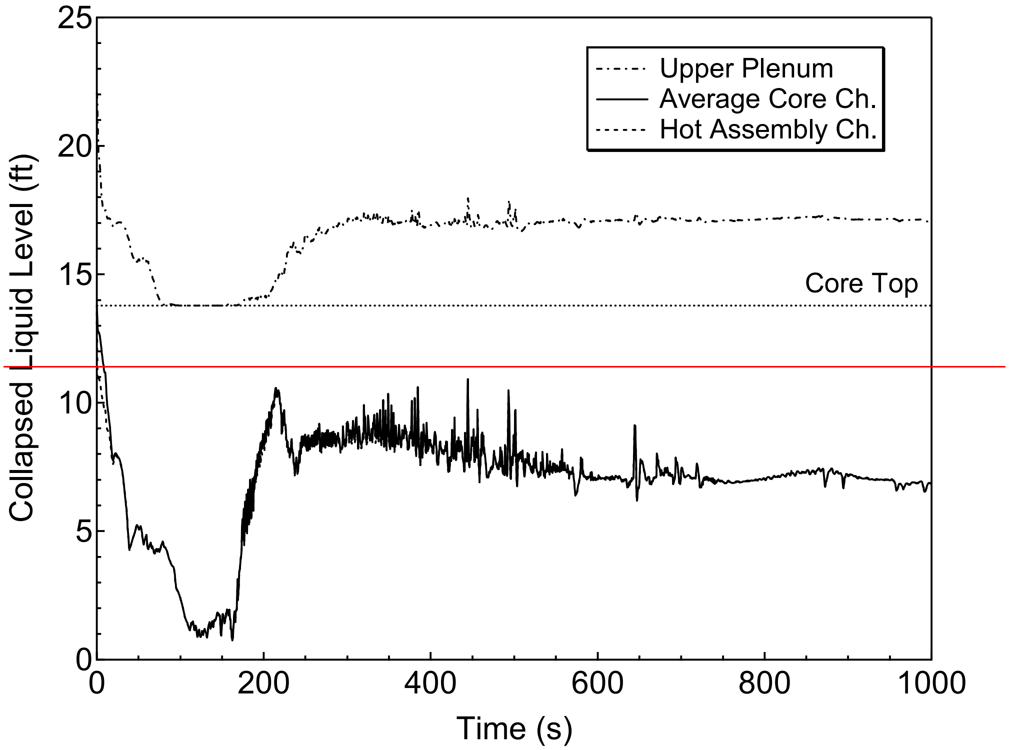


Figure 15.6.5-28 Downcomer Collapsed Level for 1-ft^2 Small Break LOCA



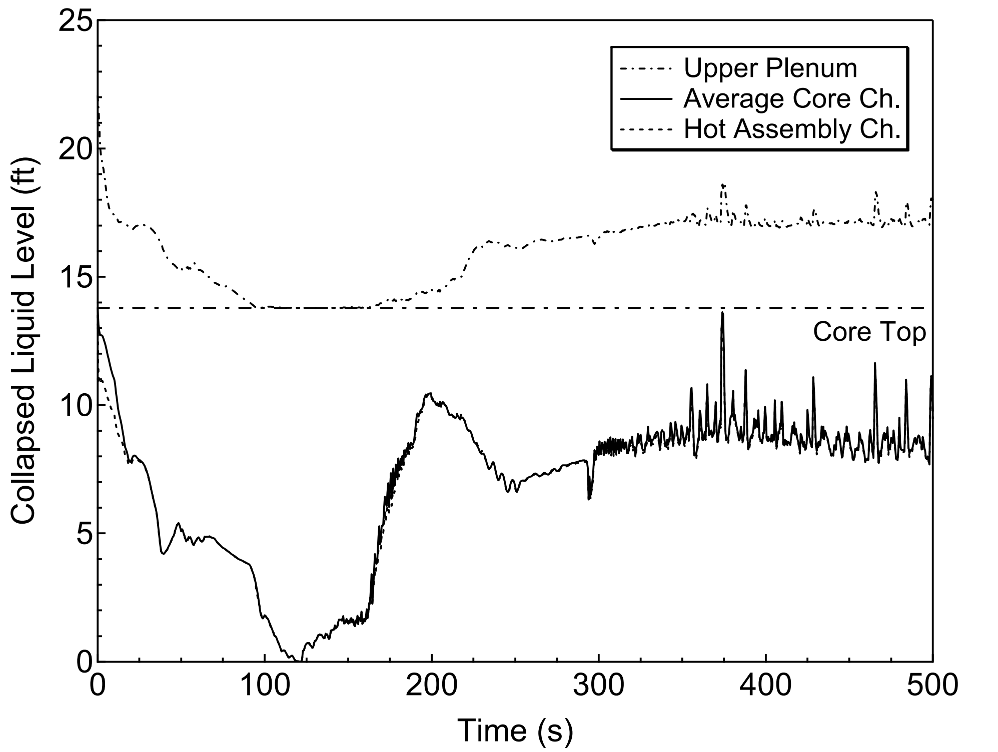


Figure 15.6.5-29 Core/Upper Plenum Collapsed Level for 1-ft^2 Small Break LOCA

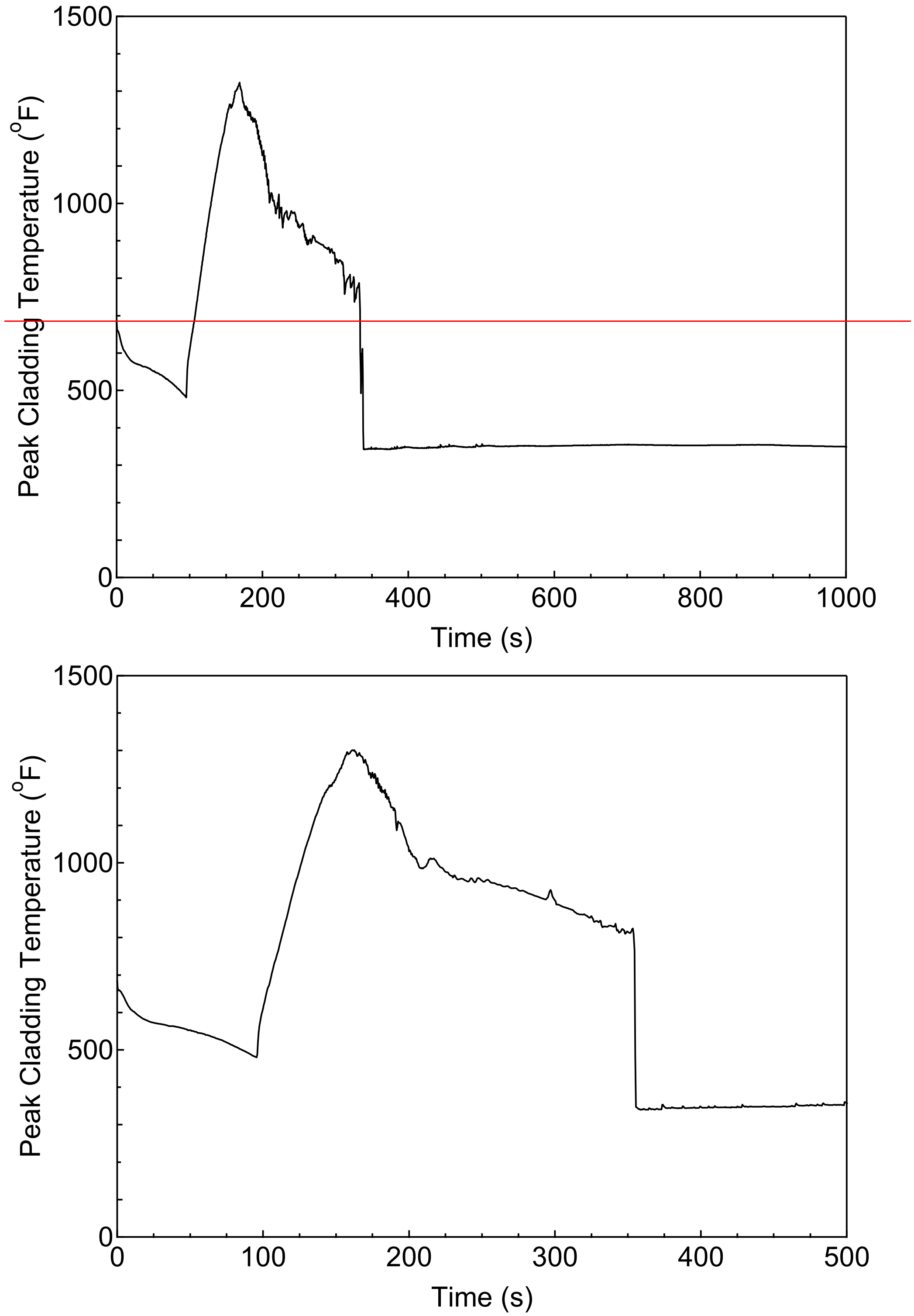


Figure 15.6.5-30 PCT at All Elevations for Hot Rod in Hot Assembly for 1-ft^2 Small Break LOCA

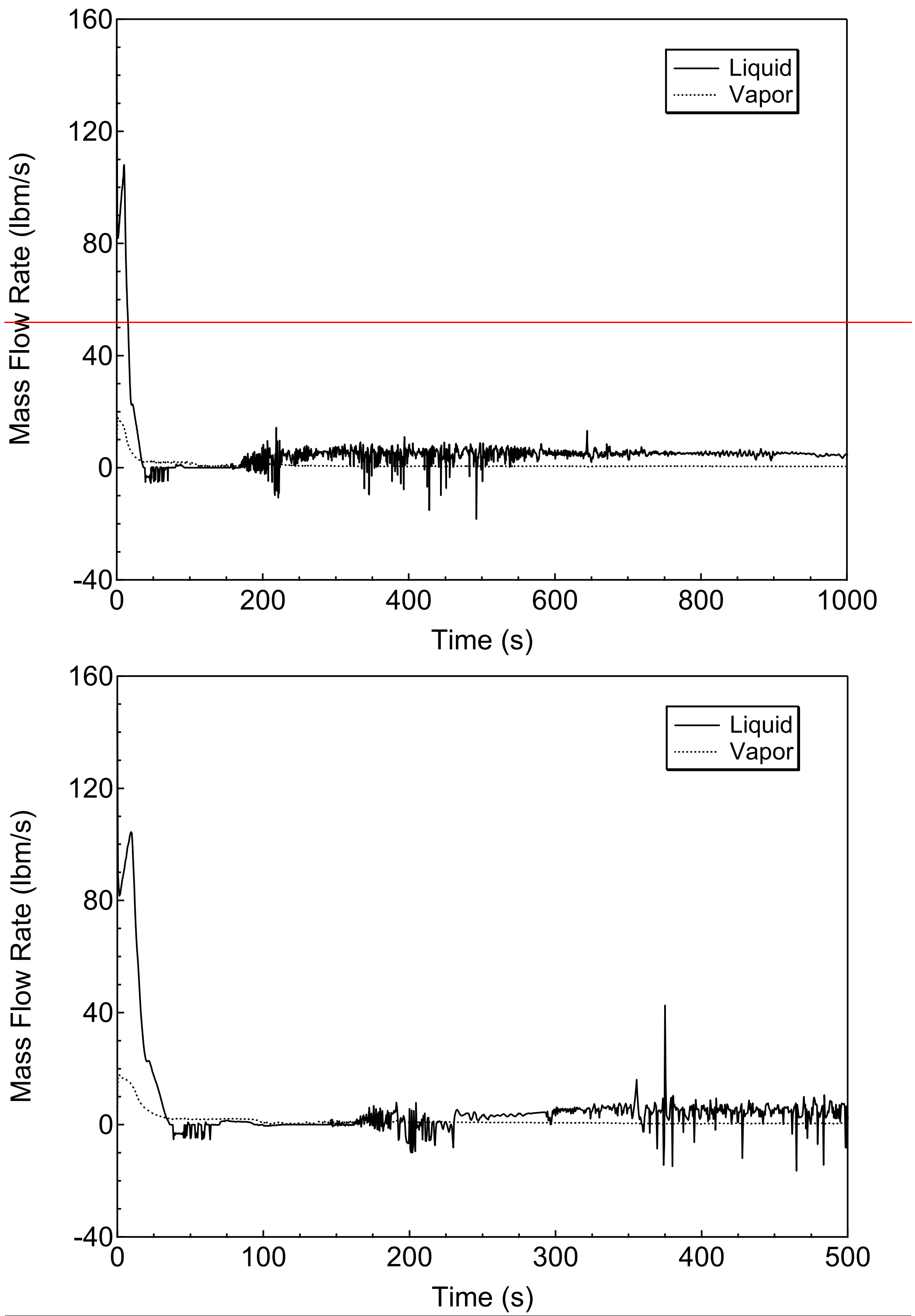


Figure 15.6.5-31 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 1-ft^2 Small Break LOCA

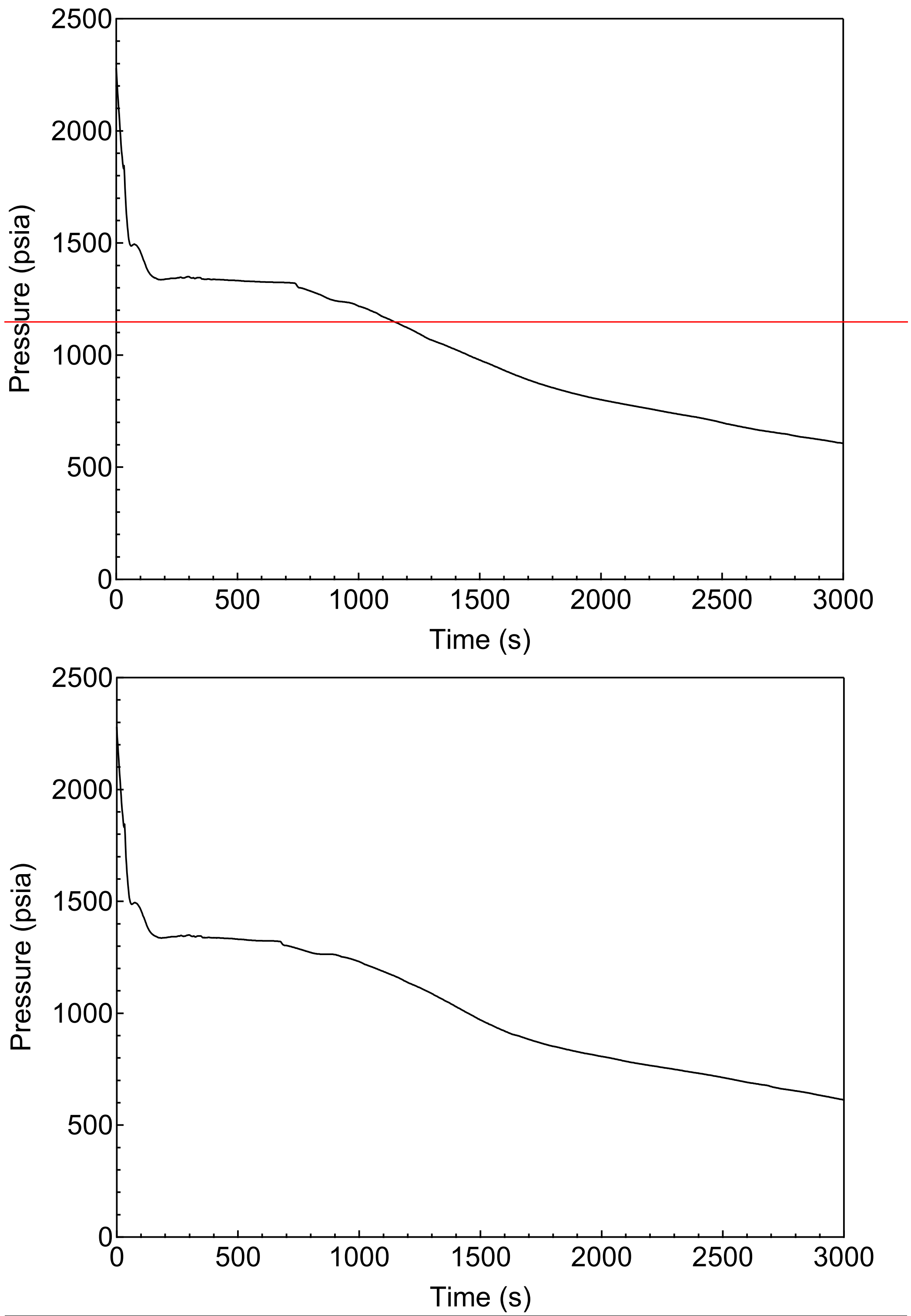
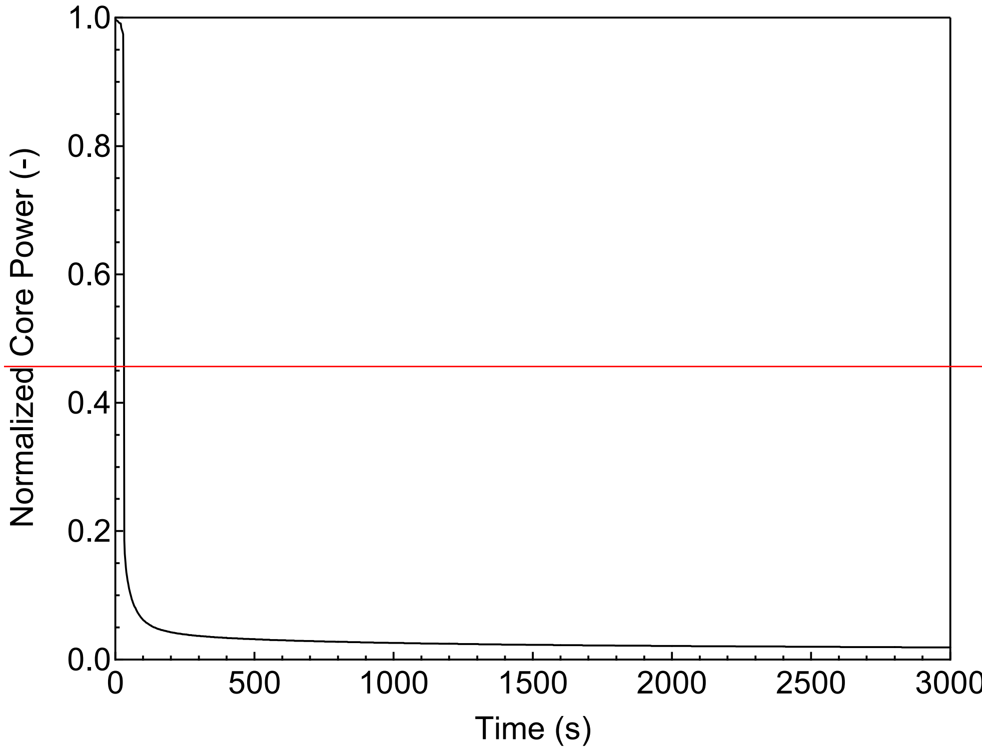


Figure 15.6.5-32 RCS (Pressurizer) Pressure Transient for DVI-line Small Break LOCA



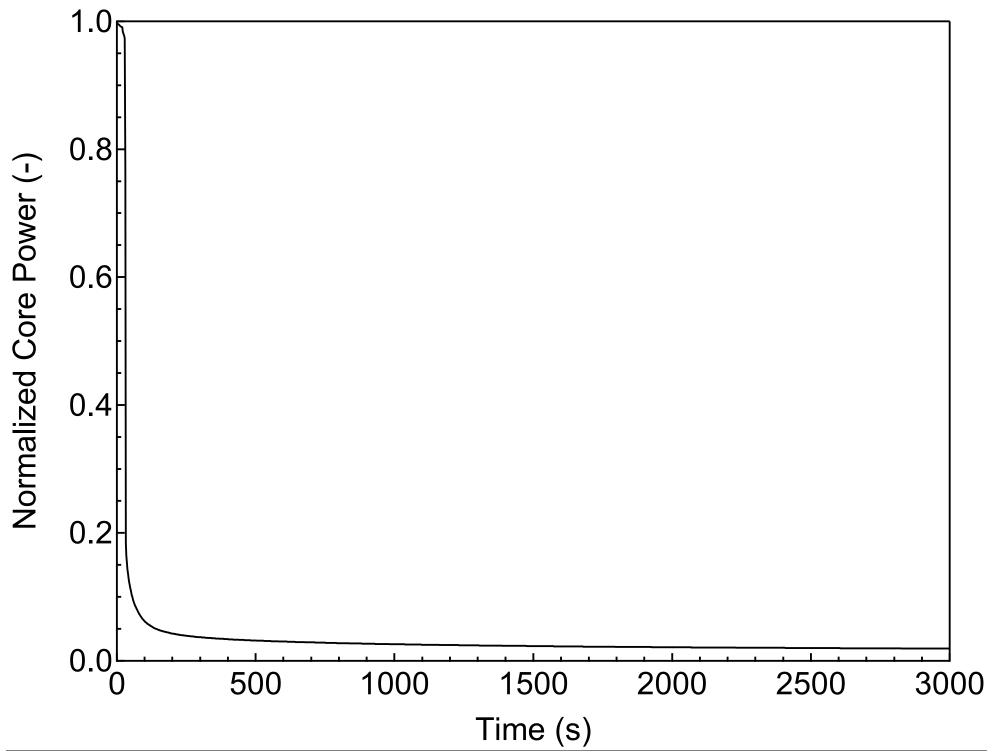


Figure 15.6.5-33 Normalized Core Power for DVI-line Small Break LOCA

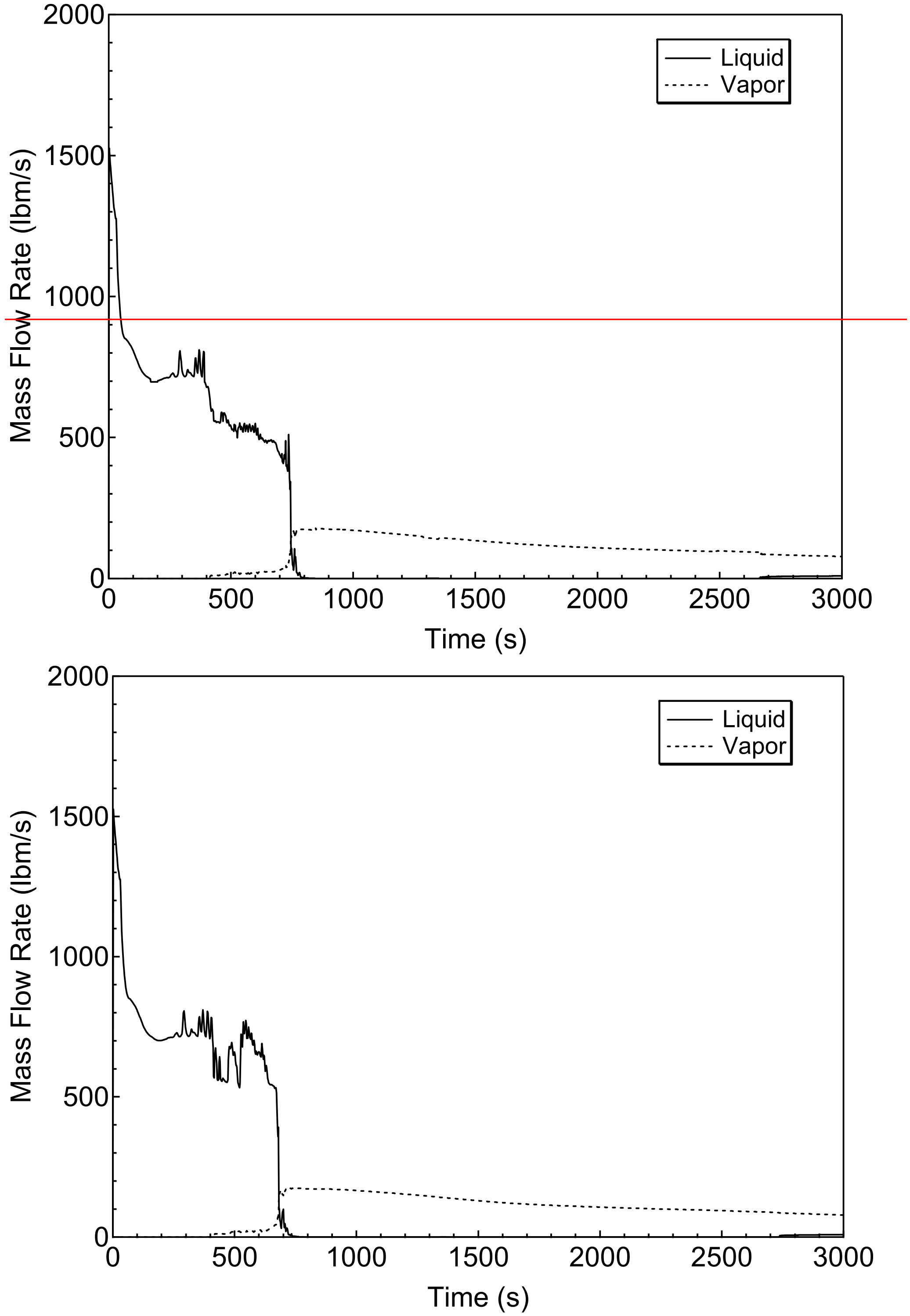
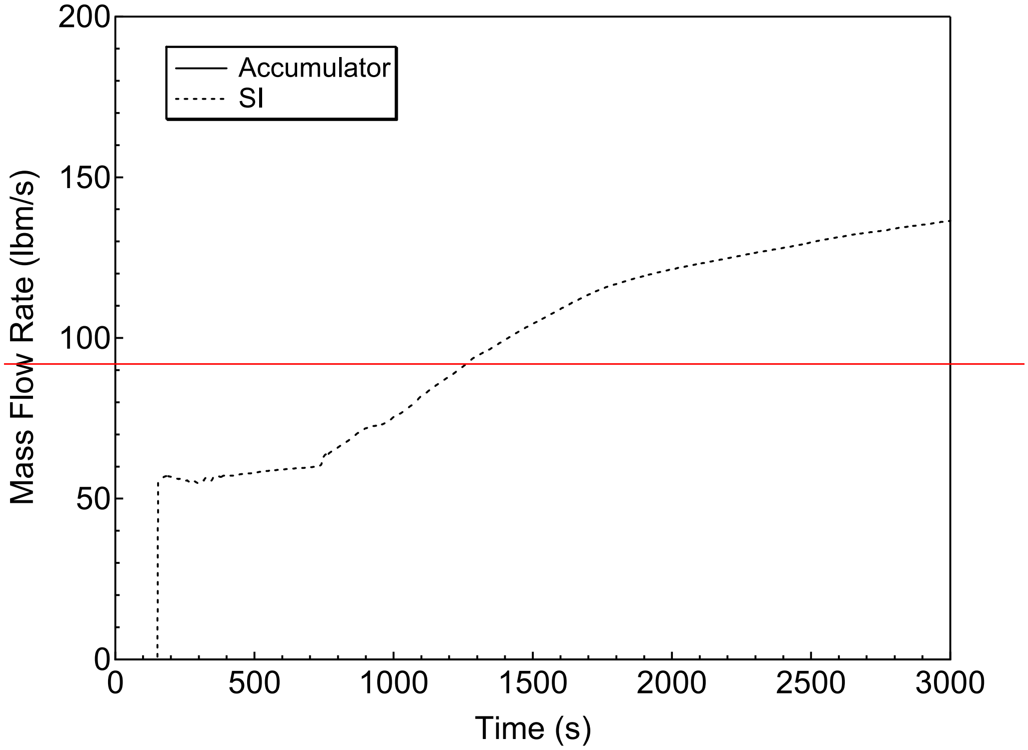


Figure 15.6.5-34 Liquid and Vapor Discharges through the Break for DVI-line Small Break LOCA



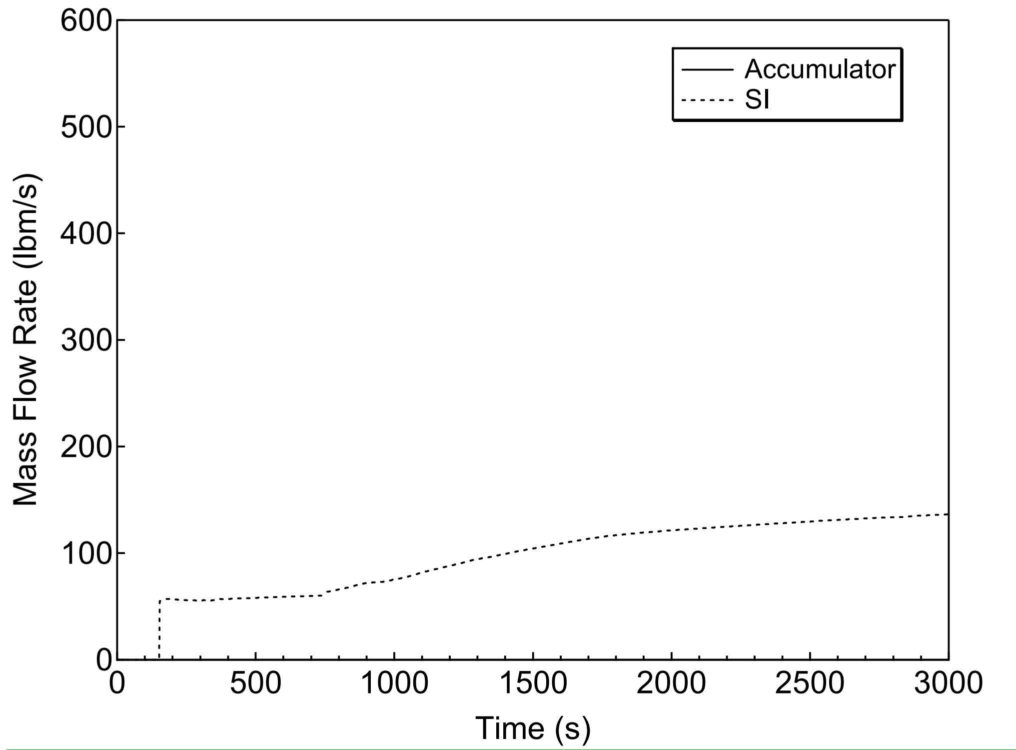
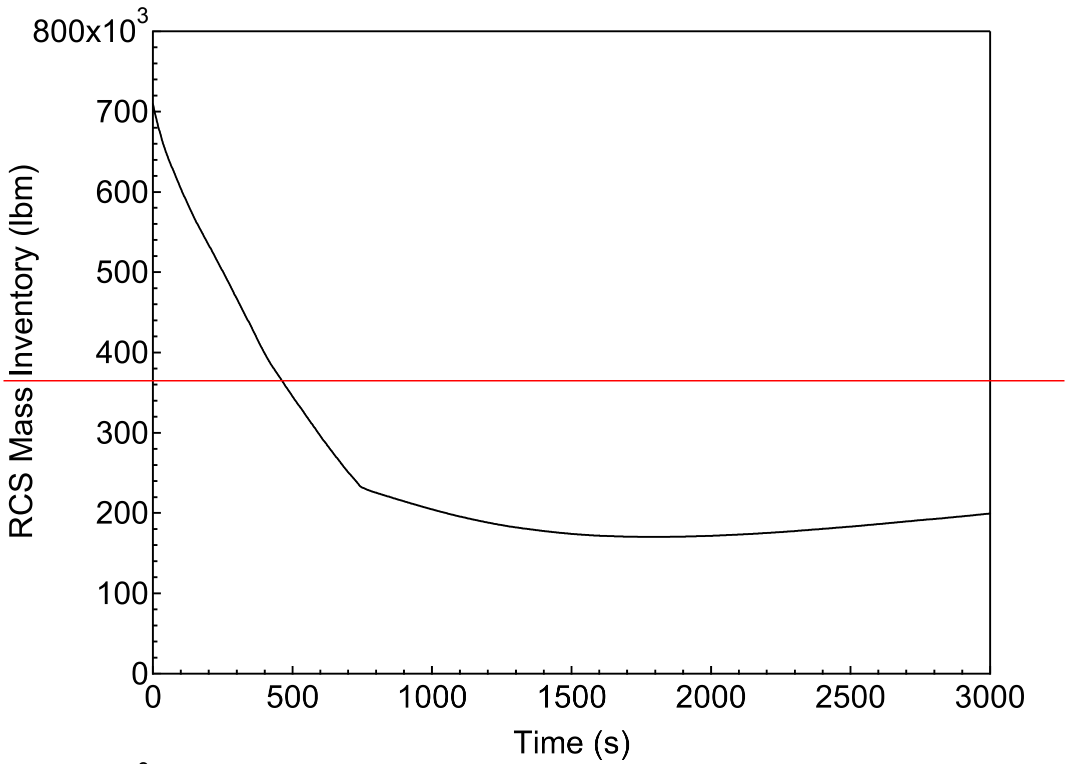


Figure 15.6.5-35 Accumulator and Safety Injection Mass Flowrates for DVI-line Small Break LOCA



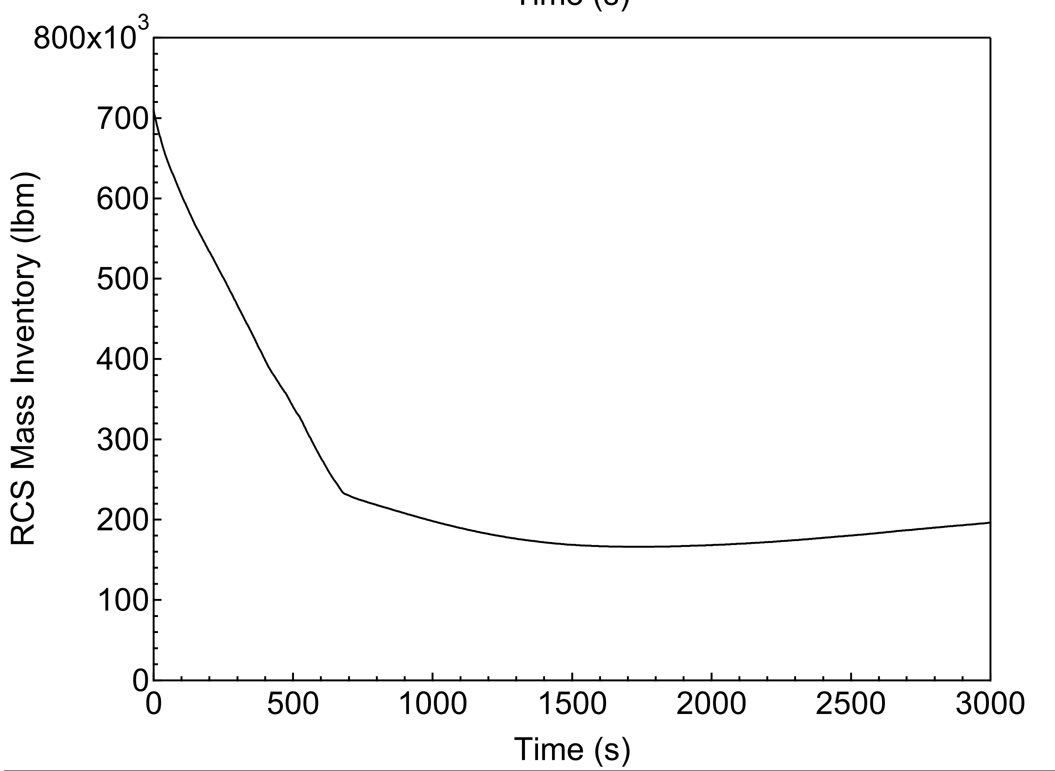
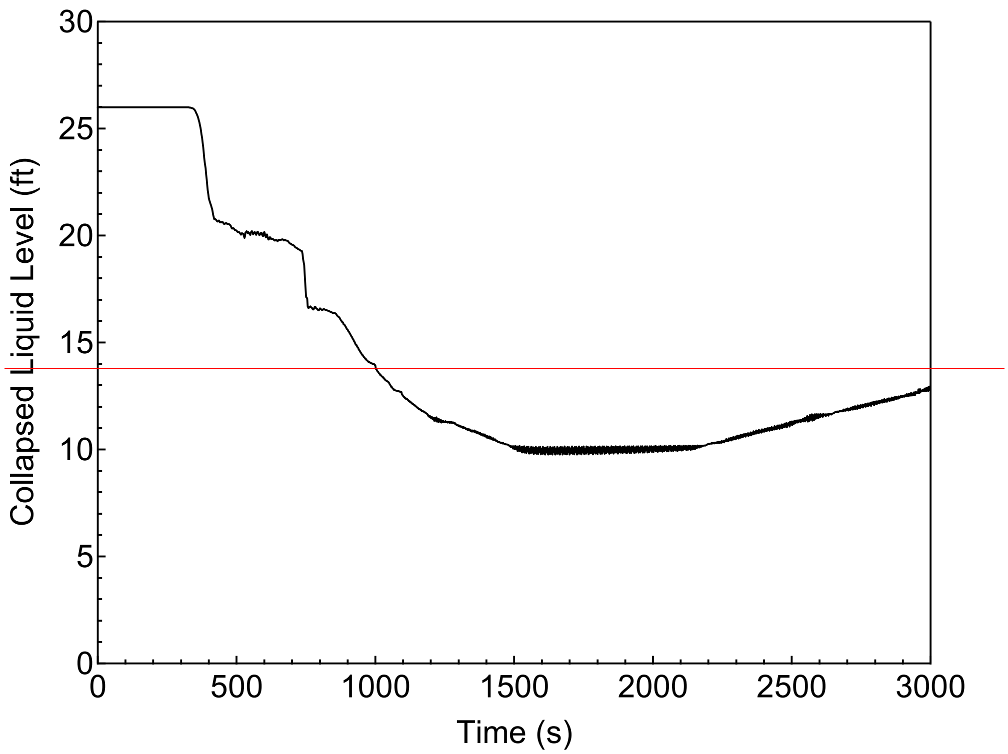


Figure 15.6.5-36 RCS Mass Inventory for DVI-line Small Break LOCA



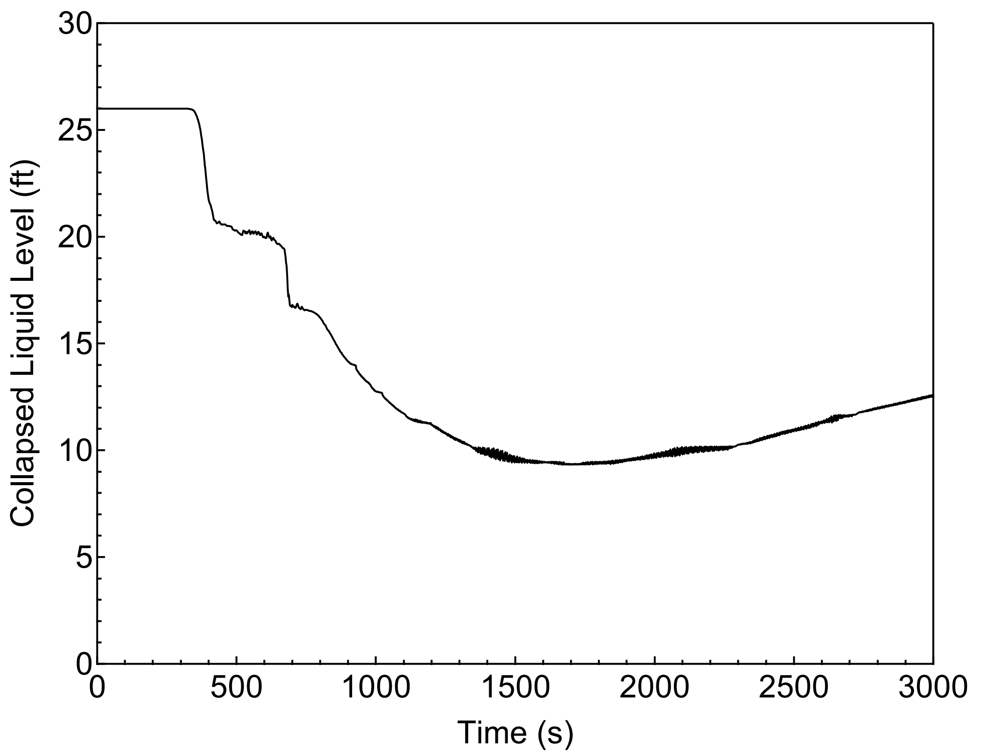


Figure 15.6.5-37 Downcomer Collapsed Level for DVI-line Small Break LOCA

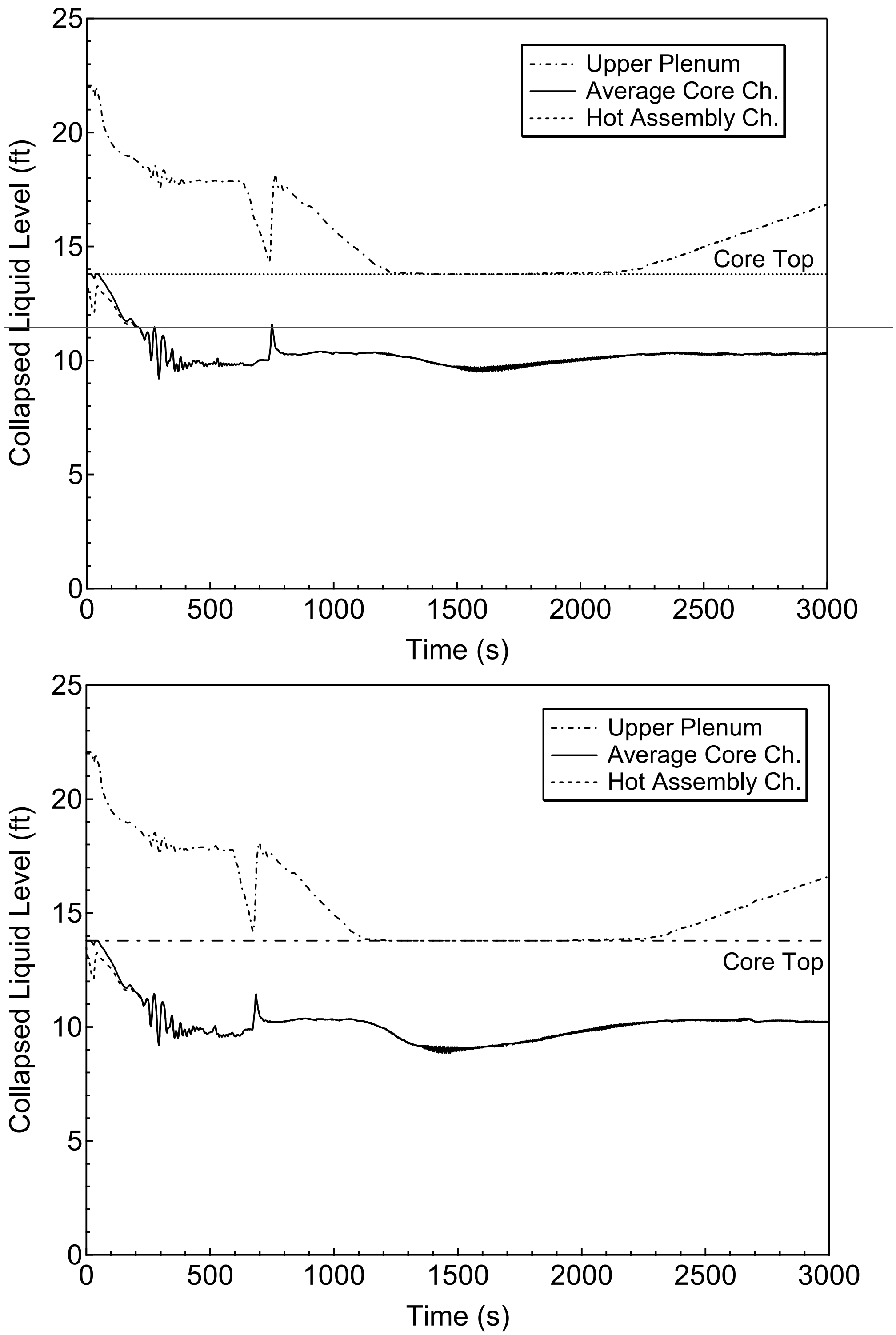


Figure 15.6.5-38 Core/Upper Plenum Collapsed Level for DVI-line Small Break LOCA

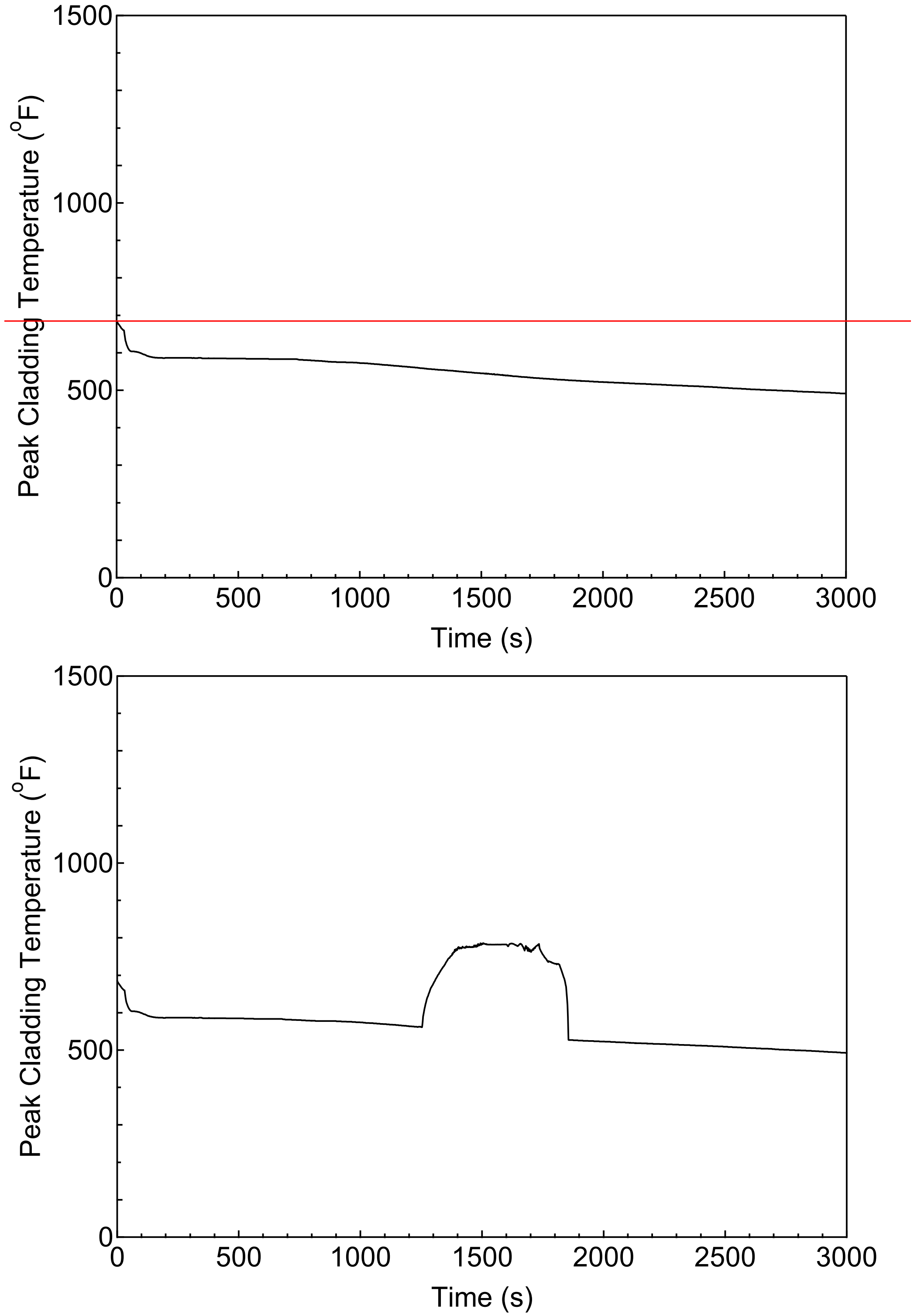


Figure 15.6.5-39 PCT at All Elevations for Hot Rod in Hot Assembly for DVI-line Small Break LOCA

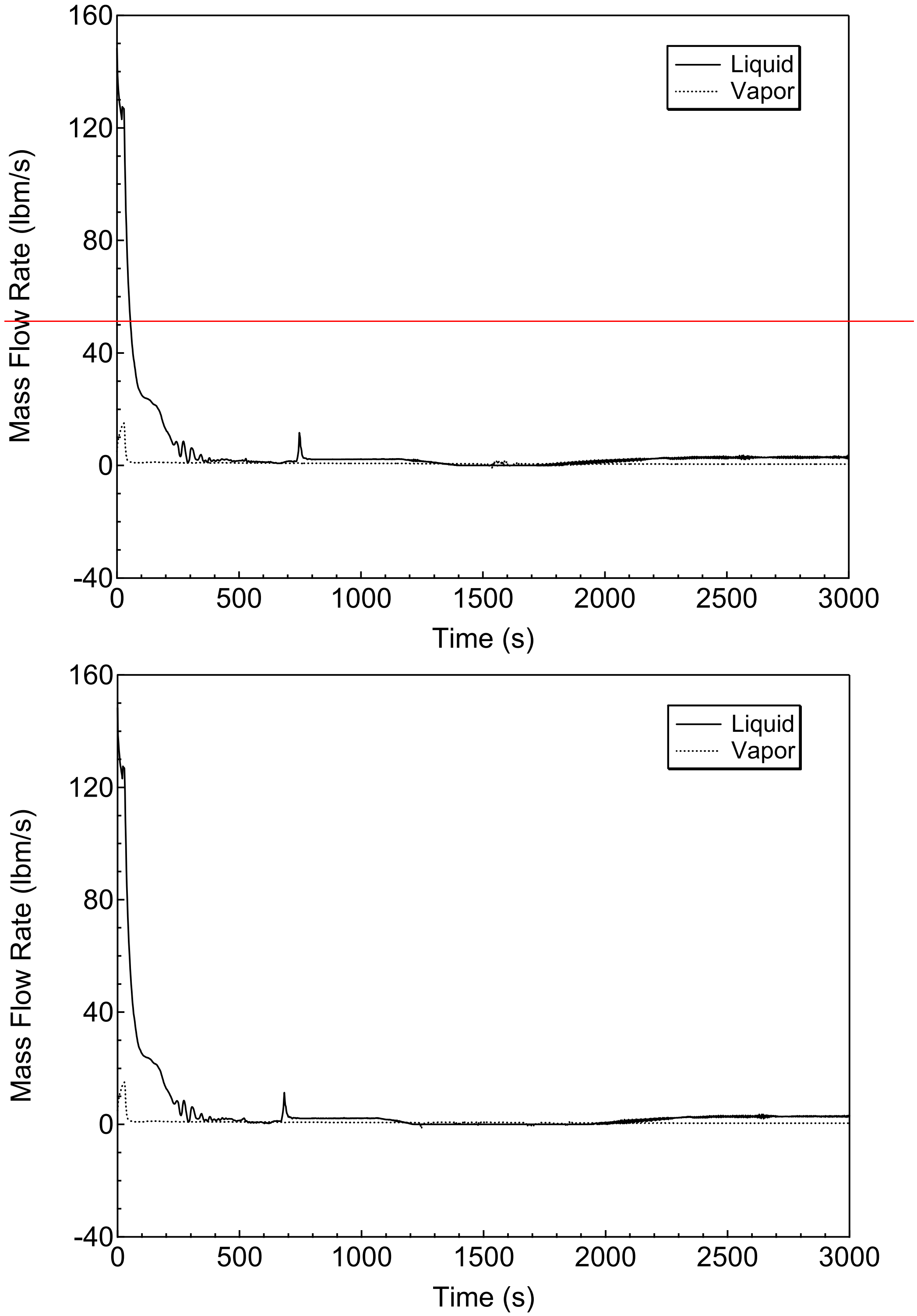


Figure 15.6.5-40 Hot Assembly Exit Vapor and Liquid Mass Flowrates for DVI-line Small Break LOCA

15.6.7 References

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15.6-22 Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Regulatory Guide 1.52, Revision 3, June 2001.

15.6-23 Laboratory Testing of Nuclear Grade Activated Charcoal, U.S. Nuclear Regulatory Commission, Generic Letter 99-02, June 3, 1999.

15.6-24 ESF Atmosphere Cleanup System, U.S. Nuclear Regulatory Commission, NUREG 0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants), Chapter 6.5.1 Revision 3, March 2007.

Radionuclides within the containment escape to the environment via leakages. The peak concentrations of radioactivity in the containment for LOCA and rod ejection accident are described in Tables 15A-15 and 15A-16.

15A.1.2 Airborne Radioactivity Removal Coefficients

The following material was taken from References 15A.5-3 and 15A.5-4, and is applicable to the US-APWR. Airborne radioactivity removal is also discussed in Section 6.5.2.

15A.1.2.1 Elemental iodine removal by wall deposition

The removal of elemental iodine by wall deposition can be estimated by the equation:

$$\lambda_w = K_w A/V$$

where:

λ_w = first-order removal coefficient by wall deposition
A = wetted-surface area
V = containment building net free volume
K_w = a mass-transfer coefficient (all available experimental data are conservatively bounded if K_w is taken to be 4.9 meters per hour)

For the US-APWR, A = 6230 m^2, V= 81000 m^3. Therefore, λ_w = 0.376 hr^{-1}.

The iodine decontamination factor (DF) is defined as the maximum iodine concentration in the containment atmosphere divided by the concentration of iodine in the containment atmosphere at some time after decontamination. The effectiveness of the wall deposition in removing elemental iodine is presumed to end when the maximum elemental iodine DF is reached. DF for the containment atmosphere achieved by the wall deposition is time dependent and is determined based on NUREG-0800, SRP 6.5.2 (Ref. 15A.5-3). The DF cannot exceed 200.

15A.1.2.2 Particulate removal

The first-order removal coefficient for particulates, λ_p~~p~~, can be determined by the method described in Reference 15A.5-3, or estimated by:

$$\lambda_p = \frac{3hFE}{2VD}$$

where:

h = spray drop fall height
V = containment building net free volume
F = spray flow
E/D = ratio of a dimensionless collection efficiency E to the average spray drop diameter D. Since the removal of particulate material chiefly depends on the relative sizes of the particles and the spray drops, it is convenient to combine parameters that cannot be known. It is conservative to assume E/D to be 10 per meter initially (i.e., 1%

Table 15A-3
Reactor Coolant Source Term (Sheet 1 of 2)

Nuclide	Half Life	Inventory (Ci)
Noble Gases		
Kr-85	10.72y	$1.24\text{x}10^{4}$
Kr-85 m	4.48h	$2.39\text{x}10^{2}$
Kr-87	76.3m	$1.55\text{x}10^{2}$
Kr-88	2.84h	$4.47\text{x}10^{2}$
Xe-133	5.245d	$4.20\text{x}10^{4}$
Xe-135	9.09h	$1.38\text{x}10^{3}$
Iodines		
I-131	8.04d	$2.16\text{x}10^{2}$
I-132	2.30h	$1.16\text{x}10^{2}$
I-133	20.8h	$3.73\text{x}10^{2}$
I-134	52.6m	$7.95\text{x}10^{1}$
I-135	6.61h	$2.44\text{x}10^{2}$
Alkali Metals		
Rb-86	18.66d	$2.25\text{x}10^{0}$
Cs-134	2.062y	$2.29\text{x}10^{2}$
Cs-136	13.1d	$6.06\text{x}10^{1}$
Cs-137	30.0y	$1.31\text{x}10^{2}$
Tellurium Group		
Sb-129	4.32h	$8.84\text{x}10^{-3}$
Te-127	9.35h	$2.72\text{x}10^{0}$
Te-127m	109d	$5.18\text{x}10^{-1}$
Te-129	69.6m	$2.20\text{x}10^{0}$
Te-129m	33.6d	$1.77\text{x}10^{0}$
Te-131m	30h	$4.68\text{x}10^{0}$
Te-132	78.2h	$5.11\text{x}10^{1}$
Strontium and Barium		
Sr-89	50.5d	$5.68\text{x}10^{-1}$
Sr-90	29.12y	$3.69\text{x}10^{-2}$
Sr-91	9.5h	$3.82\text{x}10^{-1}$
Sr-92	2.71h	$2.13\text{x}10^{-1}$
Ba-140	12.74d~~.~~	$6.96\text{x}10^{-1}$

Chapter 16

US-APWR DCD Chapter 16 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
Technical Specifications		
3.8.9-3	Table 3.8.9-1	Correction (editorial corrections) Revised “(a)” to “(1)”
3.8.9-3	Table 3.8.9-1	Editorial change Added Note ”(2) One of the two train buses may be removed from operation when switching from one train to another train”.
4.0-1	4.2.1 Fuel Assemblies	Others Replaced ”ZIRLO cladding” .with ”NRC approved cladding material”
5.2-1	5.2.1 Onsite and Offsite Organizations	Editorial change Added “the” on item a. Deleted the comma at the end of items a, b and c.
5.6-5	5.6.7 Steam Generator Tube Inspection Report	Editorial change Deleted “the” from 1st sentence.
Bases		
B 3.1.1-2	Item b	Editorial change Added “below.”
B 3.3.1-3	3rd paragraph	Letter No.: UAP-HF-11001 RAI No. 672-4982 QUESTION No. 07-02-4 Replaced “The RTS instrumentation is segmented into three distinct …” with “The RTS instrumentation is segmented into four distinct ...”.
B 3.3.1-3	3rd paragraph	Letter No.: UAP-HF-11001 RAI No. 672-4982 QUESTION No. 07-02-4 Added “4. Manual Reactor Trip switches: provide the manual reactor trip initiation in the control room.”
B 3.4.8-2	APPLICABILITY	Editorial change Move period before closing quotation marks. Remove extra (and misplaced) closing quotation marks.
B3.4.10-2	APPLICABLE SAFETY ANALYSES	Editorial: Clarification of overpressurization events. Added ”Feedwater line break” for overpressurization events.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
B3.7.10-4	APPLICABLE SAFETY ANALYSES	Reflected the additional comments of COL applicant
B 3.8.1-5	ACTIONS A.2.1 [and A.2.2]	Correction (editorial corrections) Replaced “Condition A” with “one required offsite circuit inoperable”.
B 3.8.3-1	BACKGROUND	Correction (editorial corrections) Added “the”.
B 3.8.9-3	LCO	Editorial change Added “This LCO is modified by Notes. Note 2 permits the two train buses to be removed from operation when switching from one train to another. The circumstances for de-energizing two train buses are to be limited to situations when the outage time is short.”.
16.2-1 to 2	16.2 COL 16.1_3.3.1(1) COL 16.1_3.3.2(1) COL 16.1_3.3.6(1) COL 16.1_3.4.17(1) COL 16.1_4.3.1(1) COL 16.1_5.5.9(1) COL 16.1_5.6.7(1)	Deleted Reason: Reflected the additional comments of COL applicant to be consisted with COLA.

16.2 Combined License Information

COL 16.1(1)	*Adoption of RMTS is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1(2)	*Adoption of SFCP is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1_3.3.1(1)	*~~The trip setpoints and allowable values in Table 3.3.1-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.~~Deleted.*
COL 16.1_3.3.2(1)	*~~The trip setpoints, allowable values and time delay value in Table 3.3.2-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.~~Deleted.*
COL 16.1_3.3.5(1)	*The ~~trip setpoints and~~ time delay values in SR 3.3.5.3 are to be confirmed ~~after completion of a plant specific setpoint study following selection of~~based on the plant specific ~~instrumentation~~transmission system performance.*
COL 16.1_3.3.6(1)	*~~The trip setpoints and allowable values in Table 3.3.6-1 are to be confirmed after completion of a plant specific setpoint study following selection of the plant specific instrumentation.~~Deleted.*
COL 16.1_3.4.17(1)	*~~The site specific information for tube repair is to be provided.~~ Deleted.*
COL 16.1_3.7.9(1)	*LCO 3.7.9 and associated Bases for the Ultimate Heat Sink based on plant specific design, including required UHS water volume, lowest water level for ESW pumps and maximum water temperature of the UHS, are to be developed.*
COL 16.1_3.7.10(1)	*LCO 3.7.10 and associated Bases for hazardous chemical are to be confirmed by the evaluation with site-specific condition.*
COL 16.1_3.8.4(1)	*The battery float current values in required action A.2 is to be confirmed after selection of the plant batteries.*
COL 16.1_3.8.5(1)	*The battery float current values in required action A.2 is to be confirmed after selection of the plant batteries.*
COL 16.1_3.8.6(1)	*The battery float current values in condition B, required action B.2, and SR 3.8.6.1 are to be confirmed after selection of the plant batteries.*
COL 16.1_4.1(1)	*The site specific information for site location is to be provided.*

COL 16.1_4.3.1(1)	*~~The site specific boron concentration is to be provided.~~ Deleted.*
COL 16.1_5.1.1(1)	*The titles for members of the unit staff are to be specified .*
COL 16.1_5.1.2(1)	*The titles for members of the unit staff are to be specified .*
COL 16.1_5.2.1(1)	*The titles for members of the unit staff are to be specified.*
COL 16.1_5.2.2(1)	*The titles and number for members of the unit staff are to be specified.*
COL 16.1_5.3.1(1)	*Minimum qualification for unit staff is to be specified.*
COL 16.1_5.5.1(1)	*The titles for members of the unit staff that approve the Offsite Dose Calculation Manual are to be specified.*
COL 16.1_5.5.9(1)	*~~The site specific information for tube repair is to be provided.~~ Deleted.*
COL 16.1_5.5.20(1)	*Control Room Envelope Habitability Program for hazardous chemical are to be confirmed by the evaluation with site-specific condition.*
COL 16.1_5.6.1(1)	*In case of multiple unit site, the additional information for submittal of report is to be added.*
COL 16.1_5.6.1(2)	*The format of the Annual Radiological Environmental Operating Report is to be specified based on “the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979” or another format.*
COL 16.1_5.6.2(1)	*In case of multiple unit site, the additional information for submittal of report is to be added.*
COL 16.1_5.6.7(1)	*~~The site specific information for tube repair is to be provided.~~ Deleted.*
COL 16.1_5.7(1)	*The site specific information about High Radiation Area is to be provided.*

Table 3.8.9-1 Distribution System Operating Requirements

Distribution System	Requirements	Conditions
1. 6.9 kV Class 1E, A, B, C, and D	3	A
2. 480V Load Centers, A, B, C, and D	3	A
3. 480 V Load Centers A1 and D1	2 (2)	A
4. 480V MCCs A, B, C, and D	3	A
5. 480 V MCCs A1 and D1	2 (2)	A
6. 480V MOV MCCs	4 (~~a~~1)	A
7. 120 V Vital AC Buses A, B, C, and D	4	B
8. 125 VDC Buses A, B, C, and D	3	C
9. 125 VDC Buses A1 and D1	2(2)	C
10. 125 VDC Distribution Panels	3	C

Note
(~~a~~1) For 480V MOV MCCs A and D both MCC 1 and MCC 2 are required to be OPERABLE.
(2) One of the two train buses may be removed from operation when switching from one train to another train.

4.0 DESIGN FEATURES

4.1 Site Location

[Text description of site location.]

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 257 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with ~~ZIRLO cladding~~ NRC approved cladding material, which is a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium based alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Rod Cluster Control Assemblies

The reactor core shall contain 69 Rod Cluster Control Assemblies (RCCAs) each with 24 rods per assembly. The RCCA adsorber material shall be silver indium cadmium as approved by the NRC.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout the highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the [FSAR/QA Plan]~~,~~.

b. The [plant manager] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~,~~.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~, and~~.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned for the control room from which a reactor is operating in MODES 1, 2, 3, or 4.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with ~~the~~ Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and below 230 cal/gm energy deposition for the rod ejection accident), and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution,

b. An uncontrolled rod withdrawal from subcritical or low power condition, and

BASES

BACKGROUND (continued)

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB),

2. Fuel centerline melt shall not occur, and

3. The RCS pressure SL of 2733.5 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into ~~three~~ four distinct but interconnected modules as illustrated in Chapter 7 (Ref. 2), and as identified below:

1. Field transmitters, process sensors or field contacts: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured,

2. The RPS, including Nuclear Instrumentation System (NIS): provides signal conditioning, analog to digital conversion, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to the reactor trip breakers (RTBs), and digital output to control board/control room/miscellaneous VDUs, and

3. Reactor trip breakers (RTBs): provide the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor.

4. Manual Reactor Trip switches: provide the manual reactor trip initiation in the control room.

BASES

LCO (continued)

situations when the outage time is short and core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1 is maintained or draining operations when RHR forced flow is stopped.

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other two required loops are OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE CS/RHR pump capable of providing forced flow to an OPERABLE CS/RHR heat exchanger. CS/RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level~~"~~ (MODE 6)," and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level~~"~~ (MODE 6)."~~.~~

ACTIONS

A.1

If one required RHR loop is inoperable, redundancy for RHR is lost. Action must be initiated to restore a third loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of three paths for heat removal.

B.1

If one low-pressure letdown isolation valve is inoperable, the automatic isolation function to prevent loss of RCS inventory is lost. Action must be initiated to restore the valve to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of three paths for heat removal.

BASES

APPLICABLE SAFETY ANALYSES

All accident and safety analyses that require safety valve actuation assume operation of four pressurizer safety valves to limit increases in RCS pressure. Accidents that could result in overpressurization if not properly terminated include:

a. Loss of external electrical load,

b. Loss of normal feedwater flow,

c. Reactor coolant pump shaft break,

d. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition, ~~and~~

e. ~~e.~~ Spectrum of rod cluster control assembly ejection accidents, and

f. Feedwater line break

Detailed analyses of the above transients are contained in Chapter 15 (Ref. 2). Safety valve actuation is required in events a, b, and d (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions (Ref. 2 and 3).

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The four pressurizer safety valves are set to open at the RCS design pressure 2485 psig, and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of four valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

BASES

APPLICABLE SAFETY ANALYSES

The MCRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The MCREFS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the design basis accident (DBA), fission product release presented in Chapter 15, Subsection 15.6.5.5 (Ref. 3).

The MCRATCS maintains the temperature between 73°F and 78°F.

The emergency pressurization mode of the MCRVS is assumed to operate following a DBA to provide protection from a radiological dose to the CRE occupants. The MCRVS also provides protection from smoke and hazardous chemicals to the CRE occupants. [The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 1).] The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown console. (Ref. 2).~~]~~

The worst case single active failure of a component of the MCRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCRVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCREFS trains and three of the four independent and redundant MCRATCS are required to be OPERABLE to provide the required redundancy to ensure that the system functions assuming the worst case single active failure occurs coincident with the loss of offsite power. Total system failure, such as from a loss of the required ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the CRE occupants in the event of a large radioactive release and in the equipment operating temperature exceeding limits in the event of an accident.

ACTIONS (continued)

A.2.1 [and A.2.2]

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in one required offsite circuit inoperable~~Condition A~~ for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and Class 1E GTGs are adequate to supply electrical power to the onsite Class 1E distribution system.

[Required Action A.2.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Requied Action is not applicable in MODE 4.]

The 72 hour Completion Time takes into account the capacity and capability of the remaining ac sources, a reasonable time for repairs, and the low probability of PA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable Class 1E GTG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that Class 1E GTGs in two trains are inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven emergency feedwater pumps. Two train systems, such as turbine driven emergency feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable Class 1E GTG.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Class 1E Gas Turbine Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each Class 1E Gas Turbine Generator (GTG) is provided with a storage tank having a fuel oil capacity sufficient to operate that gas turbine for a period of 7 days while the Class 1E GTG is supplying maximum post loss of coolant accident load demand discussed in Subsection 9.5.4 (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any four Class 1E GTGs is available. This onsite fuel oil capacity is sufficient to operate the Class 1E GTGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from the storage tank to the day tank by either of two transfer pumps associated with each storage tank. ~~All outside tanks, pumps, and piping are located underground.~~

For proper operation of the standby Class 1E GTGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The Class 1E GTG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated Class 1E GTG under all loading conditions. The system is required to circulate the lube oil to the gas turbine engine working surfaces and to remove excess heat generated by friction during operation. The engine oil sump in each Class 1E GTG gear boxes contains an inventory capable of supporting a minimum of 7 days of operation. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

Each Class 1E GTG has an air start system with adequate capacity for three successive start attempts on the Class 1E GTG without recharging the air start receiver(s).

APPLICABLE SAFETY ANALYSES

The initial conditions of Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) analyses in Chapter 6 (Ref. 4), and in Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E GTGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

BASES

LCO(continued)

OPERABLE ac electrical power distribution subsystems require the associated buses, load centers, and motor control centers to be energized to their proper voltages. OPERABLE dc electrical power distribution subsystems require the associated buses and distribution panels to be energized to their proper voltage from either the associated battery or charger. OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter or Class 1E transformer.
This LCO is modified by Notes. Note 2 permits the two train buses to be removed from operation when switching from one train to another. The circumstances for de-energizing two train buses are to be limited to situations when the outage time is short.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of PA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1 [and A.2]

With one Train A, B, C or D required ac bus, load center, or motor control center inoperable and a loss of function has not occurred, the remaining ac electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required ac buses, load centers, and motor control centers must be restored to OPERABLE status within 8 hours. [Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Requied Action is not applicable in MODE 4.]

Condition A worst scenario is one required train without ac power (i.e., no offsite power to the train and the associated GTGs inoperable). In this Condition, the unit is more vulnerable to a complete loss of ac power. It is, therefore, imperative that the unit operator's attention be focused on

Chapter 17

US-APWR DCD Chapter 17 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
17-v	Acronym “O-RAP”	Editorial correction: Change “operational reliability assurance program” to “reliability assurance program during the operations phase.”
17.4-1	17.4.2	“risk-significant” was added. “Risk significant SSCs” was replaced with “The risk-significant SSCs including both safety-related and non safetyrelated SSCs” Reason: Accompanied with review of Tier-1, Tier-2 revision content was reflected.
17.4-1	17.4.2 2nd line from the bottom	Editorial correction: Change “RAP (O-RAP)” to “reliability assurance program (O-RAP).”

ACRONYMS AND ABBREVIATIONS

I&C	instrumentation and control
ITAAC	inspection, test, analyses, and acceptance criteria
kV	kilovolt
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPSD	low power and shut down operation
M/D	motor driven
MCC	motor control center
MFWS	main feedwater system
MHI	Mitsubishi Heavy Industries, Ltd.
MOV	motor operated valve
MSS	main steam supply system
NESH	Nuclear Energy Systems Headquarters
NRC	U.S. Nuclear Regulatory Commission
O-RAP	reliability assurance program during the operations phase~~operational reliability assurance program~~
PAM	postaccident monitoring
PCMS	plant control and monitoring system
PRA	probabilistic risk assessment
QA	quality assurance
QAP	quality assurance program
QAPD	quality assurance program description
RAP	reliability assurance program
RAW	risk achievement worth
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RPS	reactor protection system
RRW	risk reduction worth
RTNSS	regulatory treatment of non-safety-related systems
RWAT	refueling water auxiliary tank
RWS	refueling water storage
RWSP	refueling water storage pit
RWSS	refueling water storage system

17.4 Reliability Assurance Program

This section presents the US-APWR reliability assurance program (RAP).

17.4.1 New Section 17.4 in the Standard Review Plan

As noted in Item E of SECY 95-132 (Ref. 17.4-1), an applicant for design certification should establish the scope, purpose, objective, and essential elements of an effective D-RAP and would implement those portions of the D-RAP that apply to design certification. A COL Applicant is responsible for augmenting and completing the remainder of the D-RAP to include any site-specific design information and identify the risk-significant SSCs. Once the site-specific D-RAP is established and the risk-significant SSCs are identified, the procurement, fabrication, construction, and preoperational testing can be implemented in accordance with the COL holder's D-RAP or other programs and would be verified using the inspections, test, analyses and acceptance criteria (ITAAC) process.

17.4.2 Introduction

The purposes of the US-APWR RAP are to provide reasonable assurance that: 1) the US-APWR is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for the risk-significant SSCs, 2) the risk-significant SSCs do not degrade to an unacceptable level during plant operations, 3) the frequency of transients that challenge risk-significant SSCs is minimized, and 4) the risk-significant SSCs function reliably when challenged. An additional goal is to facilitate communication between the probabilistic risk assessment (PRA), the design, and the ultimate COL activity.

The PRA evaluates the US-APWR design response to a spectrum of initiating events to ensure that plant damage has a very low probability and that risk to the public is minimized. The risk-significant SSCs including both safety-related and non safety-related SSCs~~Risk-significant SSCs~~ for the US-APWR design control document (DCD) are identified and made available to the design organization.

The US-APWR D-RAP process is implemented in several phases. Phase I, the Design Certification phase, collects system information and develops a system model. This system information and model is used as input to the design phase PRA, an operating experience review, and a review for external events. The goal of the RAP during this stage is to ensure that the reactor design meets the purposes above, through the design, procurement, fabrication, construction and preoperational testing activities and programs. The results of each of these activities are provided to an expert panel (EP) which identifies risk-significant items using probabilistic, deterministic, and other methods for inclusion in the program. Phase II, the site-specific phase, introduces the plant's site-specific information to the D-RAP process. During Phase II, the site-specific SSCs are combined with the US-APWR design SSCs into a list for the specific plant. Phase III, the last phase of the D-RAP, implements the procurement, fabrication, construction, and preoperational testing. The designer, MHI, is responsible for Phase I of the D-RAP The site-specific list of SSCs is also provided as an input to the operational phase of reliability assurance program (O-RAP), ~~RAP (O-RAP)~~ which addresses the specific plant operation and maintenance activities. The objective during this stage is to ensure that

Chapter 18

US-APWR DCD Chapter 18 Rev.2, Tracking Report Rev.7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
18-xi	Acronyms	“(SWS)” was deleted Reason: Acronym was reviewed due to the adjustment with the System code
18.2-9	Table 18.2-1	“SWS service water system” was deleted from Acronyms Reason: Acronym was reviewed due to the adjustment with the System code

ACRONYMS AND ABBREVIATIONS (Continued)

SRO	senior reactor operator
STA	shift technical advisor
SW	service water
~~SWS~~	~~service water system~~
TC	thermo couple
TSC	technical support center
V&V	verification and validation
VDU	visual display unit

Table 18.2-1 Examples of Issues and Resolutions from US-APWR OER Report (Sheet 3 of 3)

No.	Item	Issue/Scope	Human Factors Aspect Issue	Human Factors Issue addressed by US-APWR
6	GI-23	Reactor coolant pump seal failures	This is a multifaceted issue, which includes a number of proposed resolutions. One sub issue is the provision of adequate seal instrumentation to allow the operators to take corrective actions to prevent catastrophic failure of seals (see Subsection 7.3.1 for more detail).	RCP seal flow and boundary on each RCP seal are monitored and alarmed at abnormal status in MCR. RCP seal leak and rupture event is analyzed and the procedures are prepared.
7	GI-51	Improving the reliability of open cycle service water (SW) systems	The buildup of clams, mussels, and corrosion products can cause the degradation of open cycle SW systems. Added instrumentation is one means of providing operators with the capability to monitor this buildup and take corrective action before loss of system functionality occurs.	SW system has instrumentation that detects its flow degradation. The low flow alarm informs operators of service water system ~~(SWS)~~ degradation and operators can take corrective actions.

Chapter 19

US-APWR DCD Chapter 19 Rev. 2, Tracking Report Rev. 7 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
19.2-1	19.2.2	Editorial correction: “protection” was deleted
19.2-43	19.2.6.1.1	Editorial correction: “been” was added in the last paragraph
19.2-44	19.2.6.3.2	Editorial correction: “diesel” was replaced with “gas turbine”
19 A-1	Section 19 A.1	Editorial correction: Changed “cooling or spent fuel pool” to “pit” (2x)
19 A-4	Section 19A.4.4	Consistency with Aircraft Impact Assessment: Added this section on “Core Cooling Features”

19.2 Severe Accident Evaluation

This section describes the design features for the US-APWR to prevent and mitigate severe accidents in accordance with the requirements in 10 CFR 52.47(a)(23) (Reference 19.2-1). These features specifically address the issues identified in SECY-90-016 (Reference 19.2-2) and SECY-93-087 (Reference 19.2-3), which the NRC approved in related staff requirements memoranda dated June 26, 1990, and July 21, 1993, respectively, and SECY-94-302 (Reference 19.2-4) for prevention (e.g., anticipated transient without scram, mid-loop operation, SBO, fire protection, and interfacing system LOCA), for mitigation (e.g., hydrogen generation and control, core debris coolability, high-pressure core melt ejection, containment performance, dedicated containment vent penetration) and for equipment survivability.

In addition, the US-APWR design is demonstrated to satisfy the requirements of 10 CFR 52.47(a)(8) for a design certification application. In particular, this regulation invokes 10 CFR 50.34(f)(1)(i) (Reference 19.2-5) to specify that a design-specific or plant-specific PRA should be performed to seek improvements in core heat removal system reliability and containment heat removal system reliability that are significant and practical and do not excessively impact the plant.

19.2.1 Introduction

This section provides a description of the severe accident evaluation performed for the US-APWR. Specifically, Subsection 19.2.2 provides a deterministic evaluation to show how the plant severe accident preventive features would cope with specified accident conditions. Subsection 19.2.3 provides an overview of the containment design, describes severe accident progression (in-vessel and ex-vessel), and describes severe accident mitigation features. Subsection 19.2.4 addresses containment performance goals identified in SECY-93-087 and SECY-90-016, as approved by the associated U.S. NRC staff requirements memoranda. Subsection 19.2.5 describes the actions taken during the course of a postulated severe accident by the plant operating and technical staff. Finally, Subsection 19.2.6 describes how the requirement of 10 CFR 50.34(f)(1)(I) has been met.

19.2.2 Severe Accident Prevention

The purpose of this subsection is to provide a deterministic evaluation to show how the US-APWR design's severe accident preventive features act to prevent the following events:

- Anticipated transient without scram
- Mid-loop operation
- SBO
- Fire ~~protection~~
- Intersystem LOCA

For the US-APWR, an evaluation of potential design improvements, or severe accident mitigation design alternatives (SAMDAs), has been performed to meet these requirements.

19.2.6.1.2 Purpose

The purpose of this section is to provide an evaluation of SAMDAs for the US-APWR design. The approach taken is to consider the net value of a design alternative (SAMDA) as the difference between the benefit of the modification and the cost of the enhancement, with the outcome determining whether the safety benefits of the identified SAMDA outweigh the cost of incorporation in the plant design.

The cost-benefit methodology follows the current guidance for regulatory analysis contained in NUREG/BR-0184 and NUREG/BR-0058 (References 19.2-20 and 19-2-21). Industry implementation guidance (NEI 05-01, Rev. A) is applied to identify and screen SAMDAs (Reference 19.2-22). Review of potential design alternatives will consider those of current PWR plant designs, PRA information on US-APWR, and design alternatives identified by US-APWR design personnel. Both onsite and offsite costs will be included in a manner consistent with SECY-99-169 (Reference 19.2-23).

This evaluation will include a design description, estimated cost, and estimated benefit for each alternative.

19.2.6.2 Estimate of Risk for Design

The SAMDA analysis uses two distinct analyses to form the basis for the baseline design risk. The first analysis is the Level 1 and 2 PRA of the US-APWR design. The second analysis is a Level 3 PRA analysis that integrates the Level 2 source terms to quantify the consequences based on a reference site.

The CDF from at power internal events, fire and flood events is 4.4E-06 per reactor-year and from LPSD events is 2.0E-07 per reactor-year. The LRF from at power internal events, fire and flood events is 6.1E-07 per reactor-year and from LPSD events is 2.0E-07 per reactor-year. The total CDF and LRF are therefore 4.6E-06 per reactor-year and 8.1E-07 per reactor-year, respectively (Reference 19.2-17).

The MAAP code is used to develop the fission product source term corresponding to each release category. The MACCS2 code, Version 1.13.1 (Reference 19.2-24) is used in the Level 3 PRA analysis to estimate the population dose for each release category source term. In the offsite dose risk quantification, the meteorological data of the Surry site has been used as “typical”. The 50-mile population distribution data for the Surry site in the MACCS2 code sample input file has been adjusted to be in exceedance of about 80% of the U.S. nuclear plant sites, as described in NUREG/CR-2239, “Technical Guidance for Siting Criteria Development” (Reference 19.2-25). The population data and other assumptions applied are found in the Environmental Report for the US-APWR (Reference 19.2-17).

The total population dose risk is 2.7E-01 person-rem/reactor-year, and the largest contributor is from RC3 – Containment overpressure failure due to loss of heat removal (86%). The total offsite property risk is $8.9/reactor-year, with the largest contributors are: RC3 – Containment overpressure failure due to loss of heat removal (58%), RC4 – Early containment failure (20%), and RC1 – Containment Bypass (18%).

19.2.6.3 Identification of Potential Design Improvements

19.2.6.3.1 Screening method

The approach for identifying potential design improvements followed NEI 05-01, Rev. A (Reference 19.2-22). SAMDA candidates are selected primarily from two sources; one is screening from the NEI -05-01 for pressurized water reactors (PWRs, Table 14), the other is US-APWR specific candidates considering the design and insights from the CDF and population dose risk profile. The process used for SAMDA identification follows Section 5 of NEI 05-01, and resulted in the 156 SAMDA candidates.

Two phases of evaluation are performed with the first being a Phase I qualitative screening analysis following section 6 of NEI 05-01. This screening is done to eliminate SAMDAs from further consideration, and is done to reduce the number of SAMDAs for which quantitative cost analysis in a later phase (Phase II) is necessary.

19.2.6.3.2 Screening criteria

The screening criteria identified in NEI-05-01 are applied for the US-APWR design.

As the result of phase I screening, the following 10 SAMDAs are retained for Phase II analysis (Subsection 19.2.6.5). The candidate SAMDAS for Phase II analysis are:

1. Provide additional dc battery capacity (At least one train emergency dc power can be supplied more than 24 hours.)
2. Provide an additional gas turbine~~diesel~~ generator (At least one train emergency ac power can be supplied more than 24 hours.)
3. Install an additional, buried off-site power source
4. Provide an additional high pressure injection pump with independent diesel (With dedicated pump cooling)
5. Add a service water pump (Add independent train)
6. Install an independent reactor coolant pump seal injection system, with dedicated diesel (With dedicated pump cooling)
7. Install an additional component cooling water pump (Add independent train)
8. Add a motor-driven feedwater pump (With independent room cooling)
9. Install a filtered containment vent to remove decay heat
10. Install a redundant containment spray system (Add independent train)

19.2.6.4 Risk Reduction Potential of Design Improvements

APPENDIX 19A US-APWR BEYOND DESIGN BASIS AIRCRAFT IMPACT ASSESSMENT

CONTENTS

19A US-APWR Beyond Design Basis Aircraft Impact Assessment

19A.1 Introduction and Background

The design of the US-APWR takes into account the potential effects of the impact of a large commercial aircraft, which the NRC has determined is a beyond design basis event. In accordance with 10 CFR 50.150(a), a design-specific assessment has been performed for the US-APWR using realistic analysis to demonstrate that, in the event an US-APWR is struck by a large commercial aircraft, design features and functional capabilities exist to ensure that the following functions are maintained:

- The reactor core remains cooled, or the containment remains intact; and
- Spent fuel pit ~~cooling or spent fuel pool~~ integrity is maintained.

The assessment demonstrates the inherent robustness of the US-APWR design with regard to potential large aircraft impacts.

Specific assumptions used in the US-APWR aircraft impact assessment are based on requirements and guidance provided by the NRC and the Nuclear Energy Institute (NEI). The NRC provided the physical characteristics, including the loading function of the impacting aircraft, in July of 2007 (Reference 19A-1). The methodology for assessing effects for aircraft impact are described in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 7 (Reference 19A-2).

This appendix describes the design features and functional capabilities of the US-APWR identified in the detailed assessment that assure the reactor core remains cooled or the reinforced concrete containment vessel (PCCV) remains intact, and spent fuel pit ~~cooling or spent fuel pool~~ integrity is maintained. These identified design features are designated as "key" design features. ~~and functional capabilities.~~

19A.2 Scope of the Assessment

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.3 Assessment Methodology

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.4 Assessment Results

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.4.1 PCCV

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.4.2 Plant Arrangement

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.4.3 Fire Barriers and Fire Protection Features

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.4.4 Core Cooling Features

Security-Related Information – Withheld Under 10 CFR 2.390

(SRI)

19A.5 Conclusions of Assessment

This assessment concludes that key design features and functional capabilities of the US-APWR ensure adequate protection of public health and safety in the event of an impact of a large commercial aircraft, as defined by the NRC. The postulated aircraft impacts would not impair the US-APWR's core cooling capability, containment integrity, or spent fuel pit integrity~~, or adequate spent fuel cooling~~. The assessment resulted in identification of key design features and functional capabilities described in Section 19A.4, changes to which are required to be controlled in accordance with 10 CFR 50.150(c).

19A.6 References

1. Letter from D. Matthews, NRC to Dr C. K. Paulson, Mitsubishi Nuclear Energy Systems, Inc, Subject: "Approval of Mitsubishi Nuclear Energy Systems Safeguards Protection Program and Reviewing Official, and Transmittal of Beyond Design Basis, Large Commercial Aircraft Characteristics Specified by Commission," December 7, 2007.

2. NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 7, May 2009.

~~3.UAP-SGI-09001, " US-APWR Design Certification Aircraft Impact~~

~~Assessment," April 2009, Mitsubishi Heavy Industries, LTD.~~